How a Nuclear Reactor Works

Jeffrey A. Mahn
Nuclear Engineer (Retired)
Albuquerque, NM USA
jamahn47@gmail.com

It’s Complicated

Occasionally the question is asked, how does a nuclear reactor work.

- Simple explanation: You put lots of nuclear fuel in the reactor vessel and then control the fission reaction (heat generation rate) with neutron absorbing control rods.
- Reality: It’s complicated. It’s all about neutrons – how they are generated and how the reactor neutron population is controlled through proper reactor design and operation.

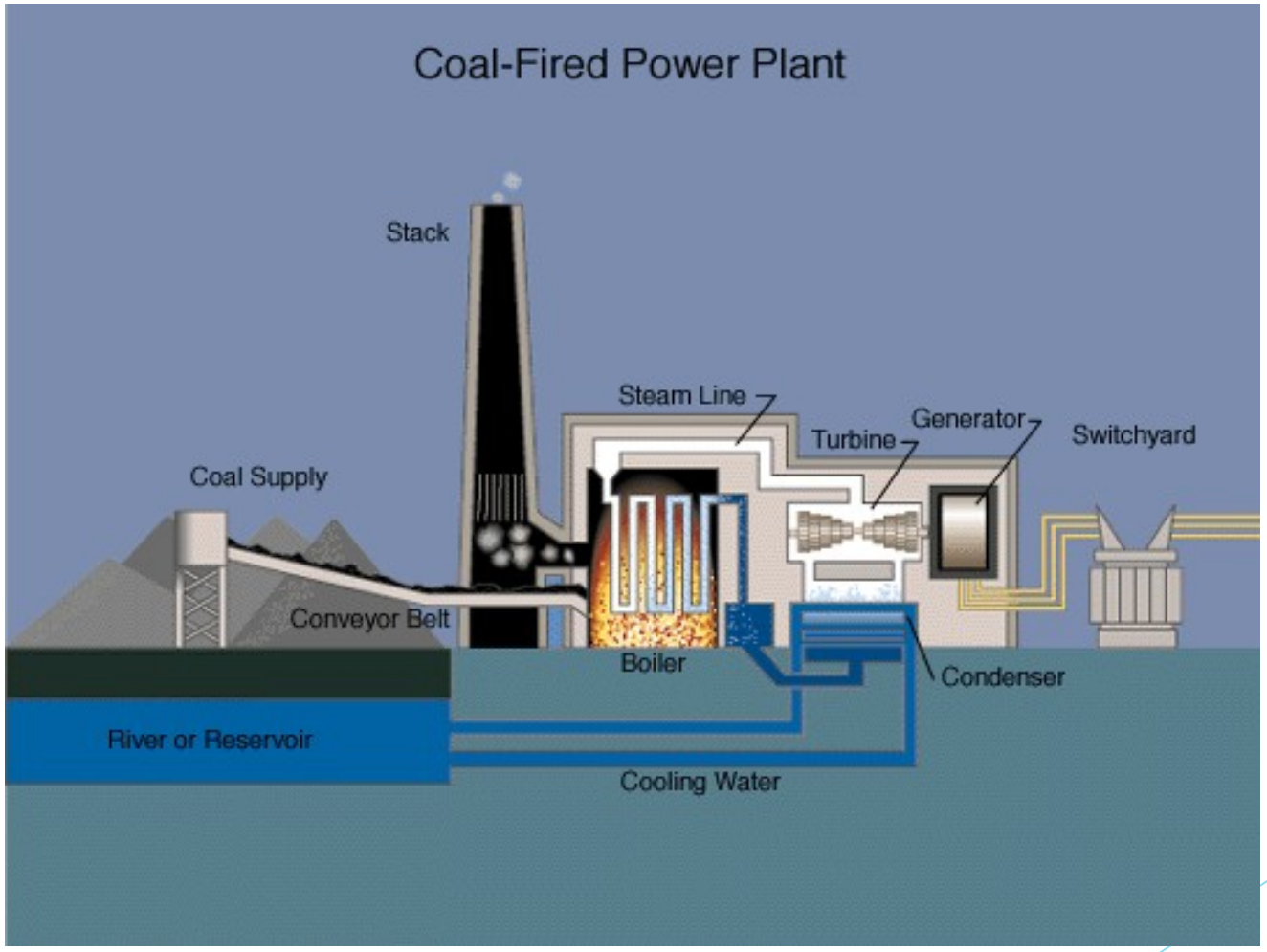
Terminology

- Curie (Ci) – quantity of any radioactive material decaying at rate of 3.7×10^{10} disintegrations (atoms) per second
- Barn – unit used to express atomic cross-sections for neutron interactions (= 10^{-24} cm^2)
- Fast neutrons, fast fission – neutron energy of ≈1 MeV
- Thermal neutrons, thermal fission – neutron energy of ≈0.025 eV (in approximate equilibrium with thermal motion of surrounding materials)
- Neutron multiplication factor (k)

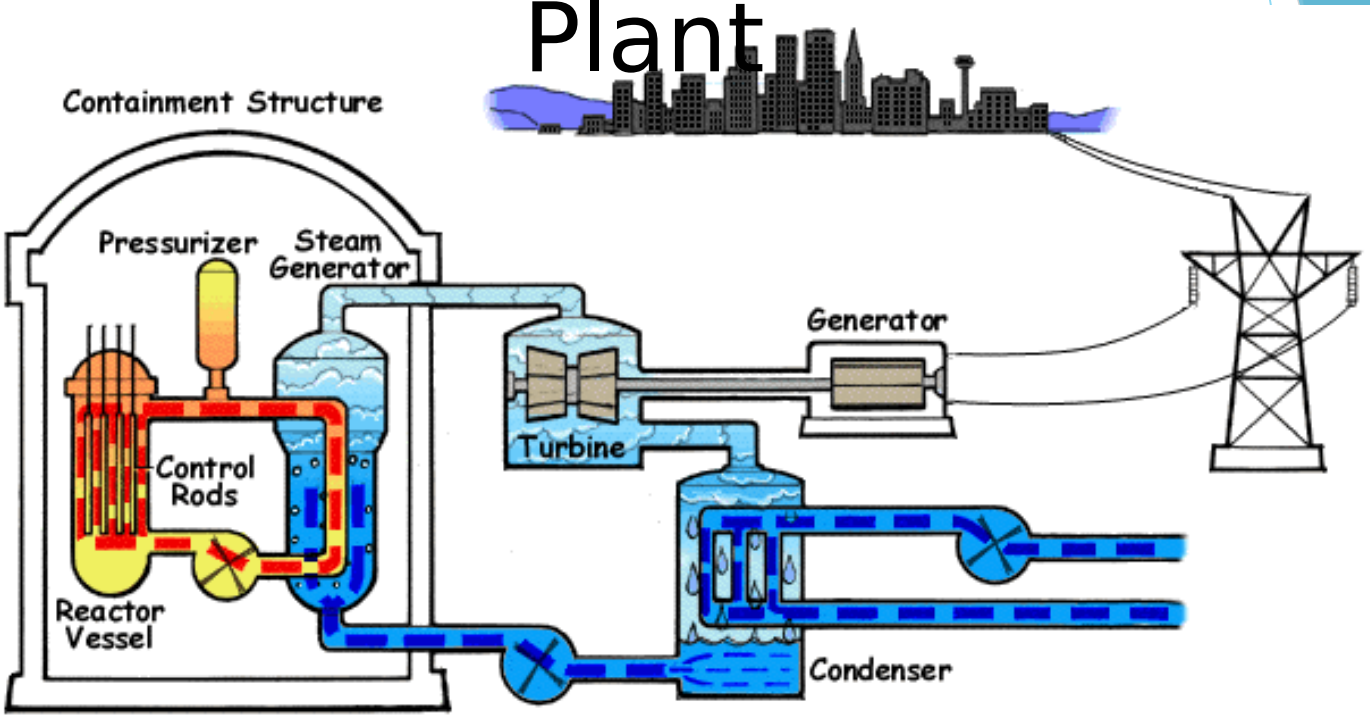
Terminology (cont.)

- Reactivity ($\rho = (k - 1)/k$) – fractional excess neutron multiplication
- Control rod – reactor core component containing neutron absorbing material
- Delayed neutron fraction (β)
- Neutron generation time – time required for neutrons from one generation to cause fissions that produce next generation of neutrons
- Reactor period (T) – time required for reactor power to change by factor of 2.718
- Subcritical multiplication factor ($M = 1/(1- k)$)

Fossil/Nuclear Power Plant Operational Comparison


Coal-Fired Power Plant
Stack
Steam Line
Turbine
Generator
Switchyard
Coal Supply
Conveyor Belt
Boiler
Condenser
River or Reservoir
Cooling Water

Pressurized Water Reactor Plant



(Animation)

Power Plant Operation Comparison

	Fossil-fueled Power Plant	Nuclear Power Plant
Heat Source Fuel Supply	Fuel fed continuously into boiler and ignited by a flame (burner)	Each reactor core fuel loading contains sufficient uranium fuel for 1-2 years of operation
Heat Generation Rate Control	Heat generation controlled by fuel feed rate	Fission reaction rate controlled by neutron absorbing materials inserted into reactor core
Means of Heat Removal	Water flowing through tubes in boiler wall	Water flowing through coolant channels in nuclear fuel assemblies
Heat Generation Termination	Termination of fuel feed to boiler	Termination of fission process by full insertion of neutron absorbing materials into reactor core*

* Radioactive decay of fission products in reactor fuel continues to generate
significant heat, which must be removed, following reactor shutdown

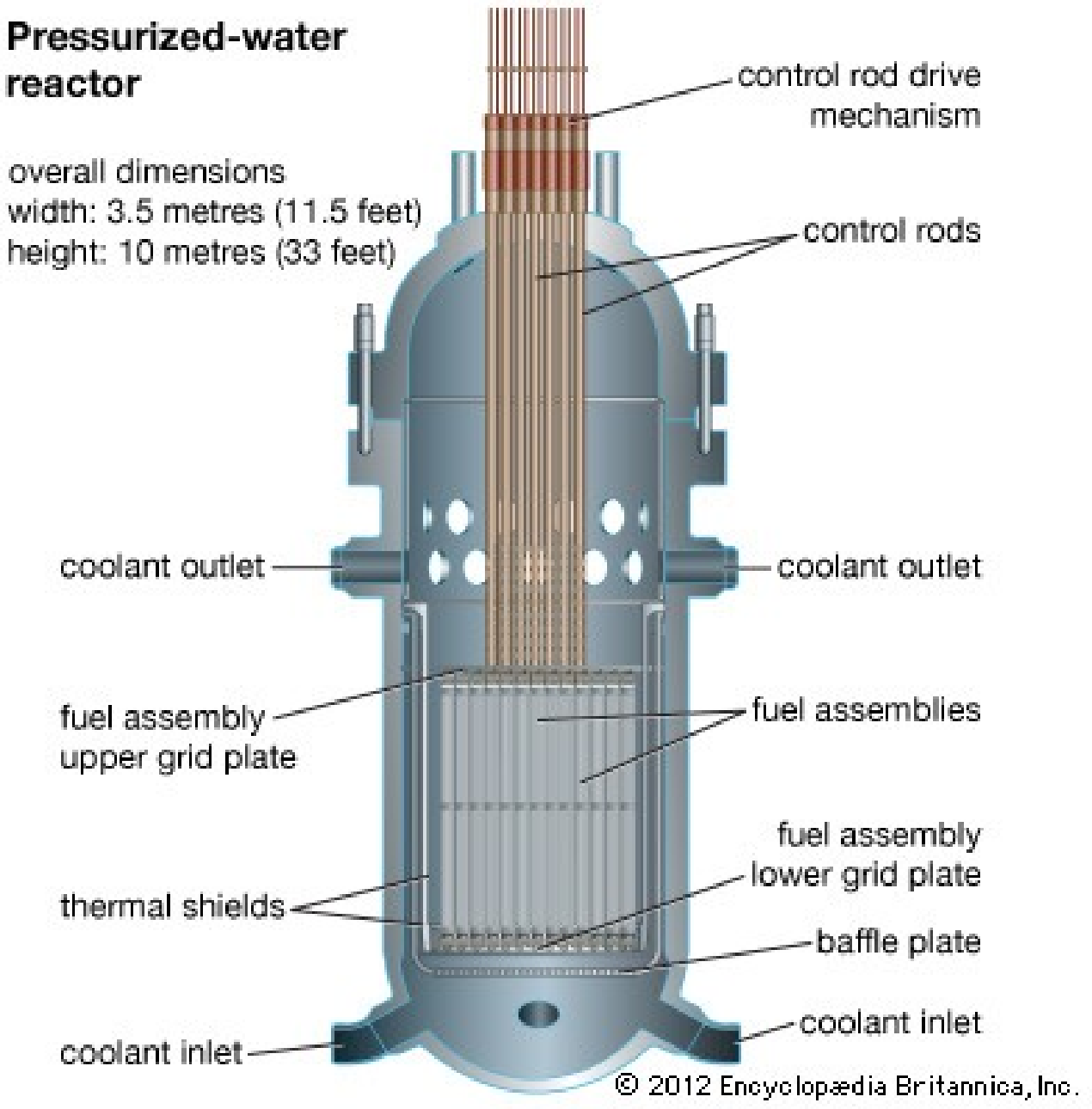
Nuclear-Generated Power

- Compared to fossil-fuel boiler, process for generating heat in nuclear reactor is unique
  - Heat generation rate dependent on fission reaction rate in nuclear reactor fuel
    - Fission reaction rate in nuclear fuel assemblies composing reactor core dependent on total neutron population at any given instant as well as rate of neutron population change
    - Drastic changes in fission neutron population (i.e., reactor power level) can occur on extremely short time scales if not properly controlled

Nuclear-Generated Power (cont.)

- Operating reactor neutron economy controlled by integration of nuclear physics and reactor design features to assure
  - enough neutrons for sustained fission chain reaction
  - sufficient total neutron population for fissions necessary to achieve desired heat generation rate (i.e., reactor power level)
  - sufficient control of neutron population changes to prevent unmanageable increases in reactor power level

Light Water Reactor (LWR) Components


Pressurized-water reactor
overall dimensions
width: 3.5 metres (11.5 feet)
height: 10 metres (33 feet)
control rod drive mechanism
control rods
coolant outlet
coolant outlet
fuel assembly upper grid plate
fuel assemblies
fuel assembly lower grid plate
thermal shields
baffle plate
coolant inlet
coolant inlet
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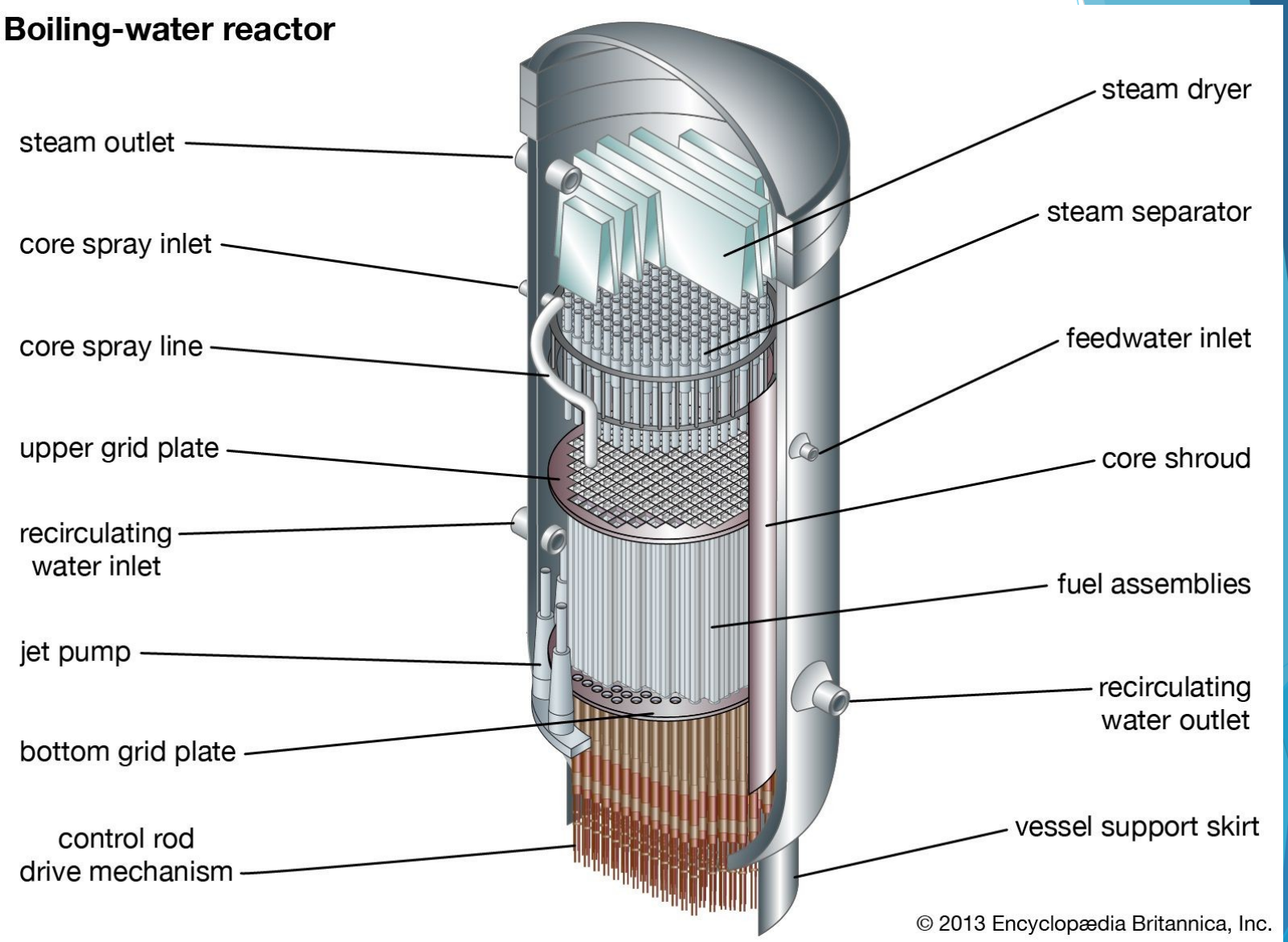
Boiling-water reactor
steam outlet
core spray inlet
core spray line
upper grid plate
recirculating water inlet
jet pump
bottom grid plate
control rod drive mechanism
steam dryer
steam separator
feedwater inlet
core shroud
fuel assemblies
recirculating water outlet
vessel support skirt
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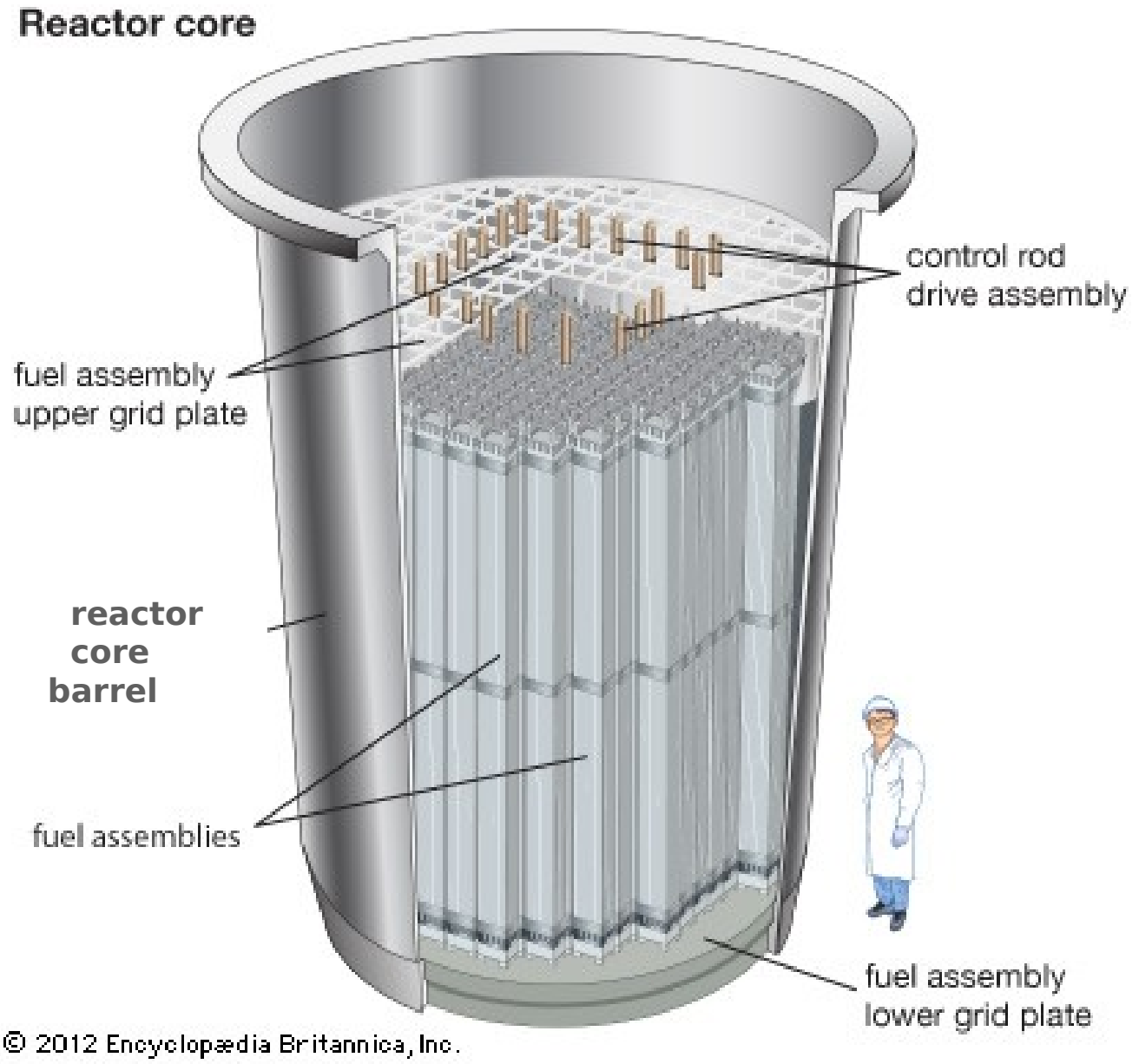
Reactor core
control rod drive assembly
fuel assembly upper grid plate
reactor core barrel
fuel assemblies
fuel assembly lower grid plate
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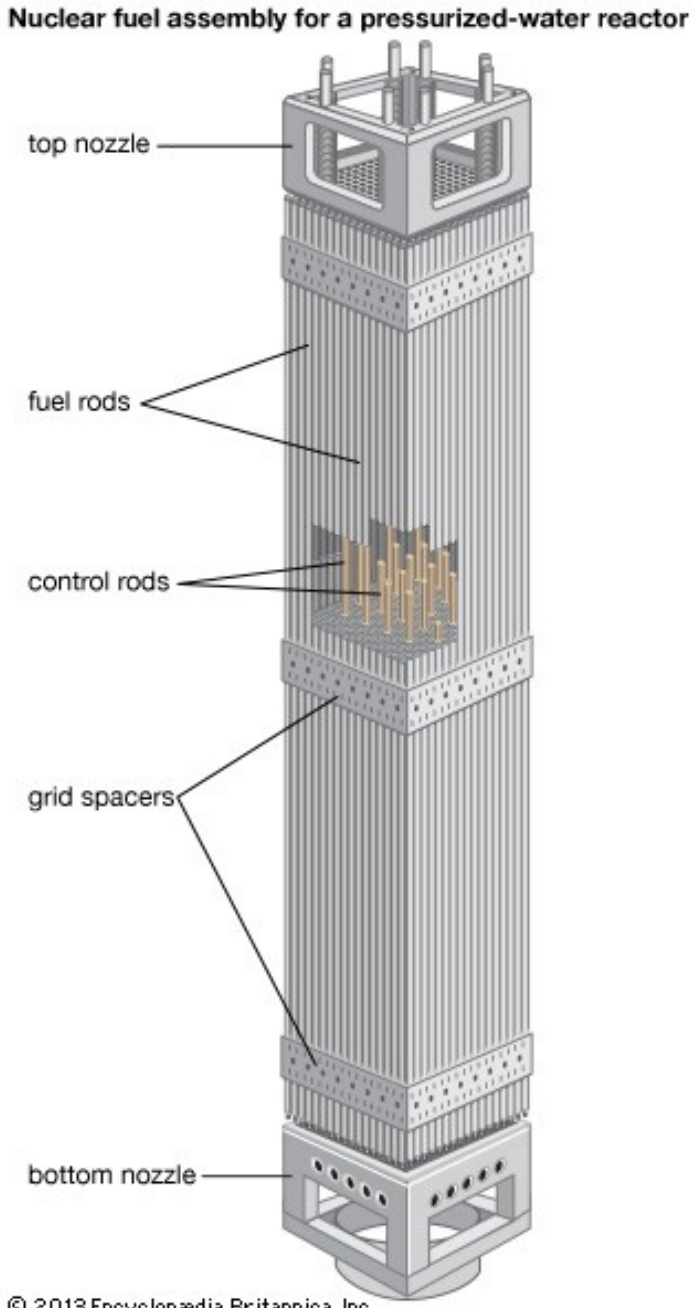
Fuel assemblies in reactor pressure vessel form reactor *core* wherein fission process produces heat energy in reactor fuel.

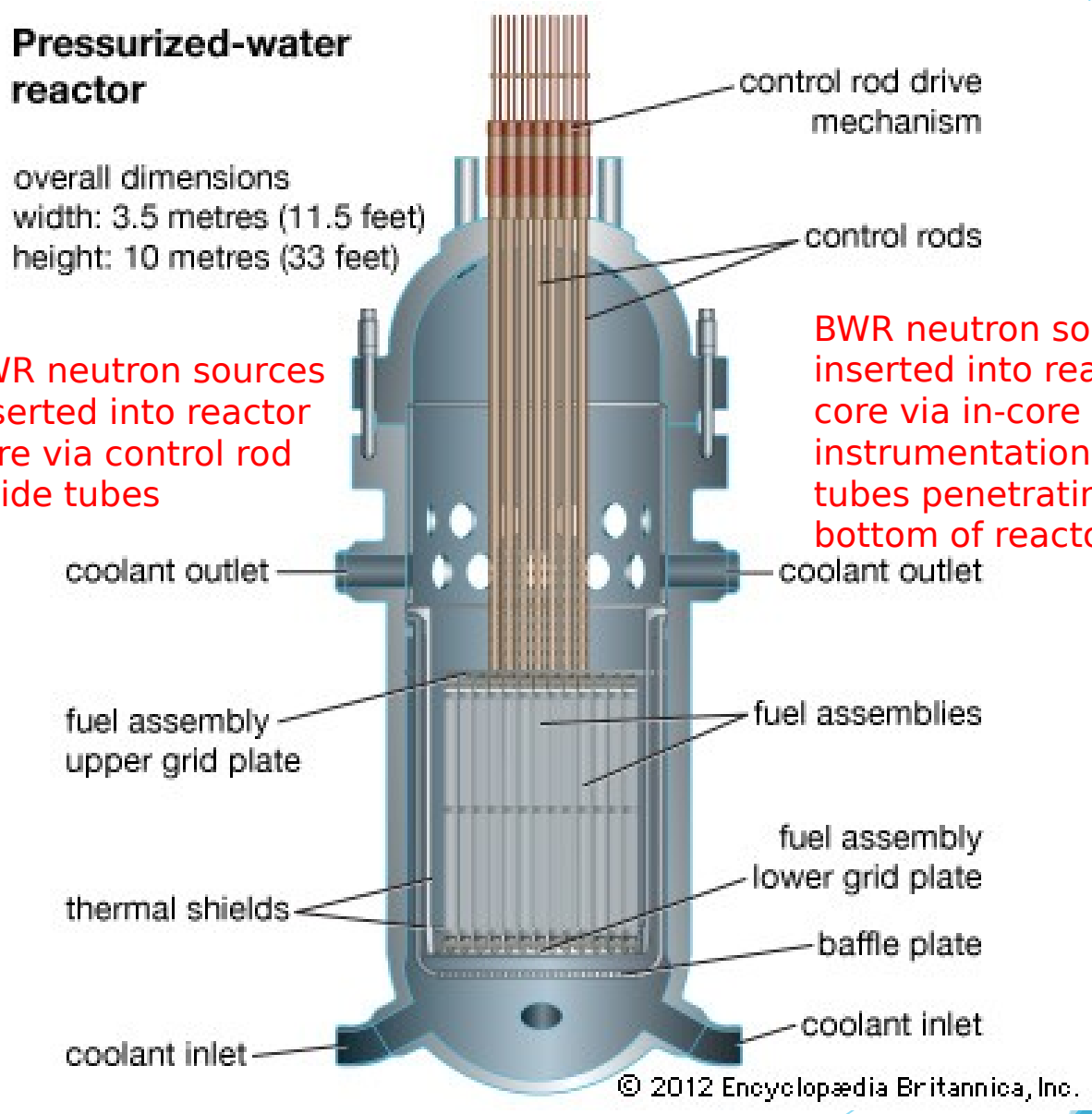


Source of Reactor Neutrons

Initial Source of Reactor Neutrons

- Radioisotopes which decay by alpha emission and collocated with beryllium can eject neutrons from beryllium nuclei
  - Americium-beryllium (≈2.3x10^6 neutrons/sec per Ci*)
  - Radium-beryllium (≈1.5x10^7 neutrons/sec per Ci)
- Spontaneous fission of Californium-252 (97% alpha decay, 3% spontaneous fission)
  - High yield per unit mass (≈4.4x10^9 neutrons/sec per Ci)
- Neutron source inserted into reactor core prior to reactor startup with fresh nuclear fuel or when restarted following long shutdown

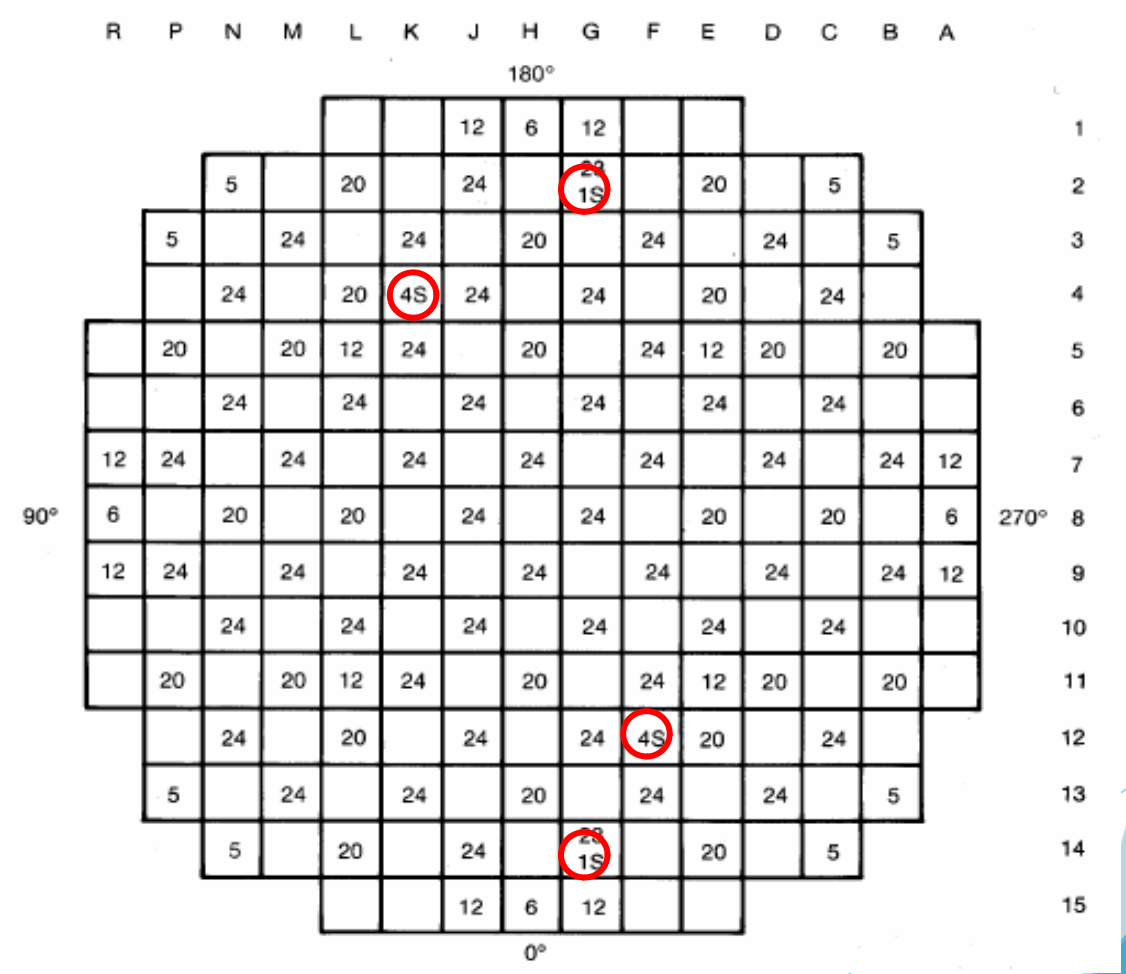
* One Curie (Ci) is equal to 3.7x10^{10} radioactive decays per second



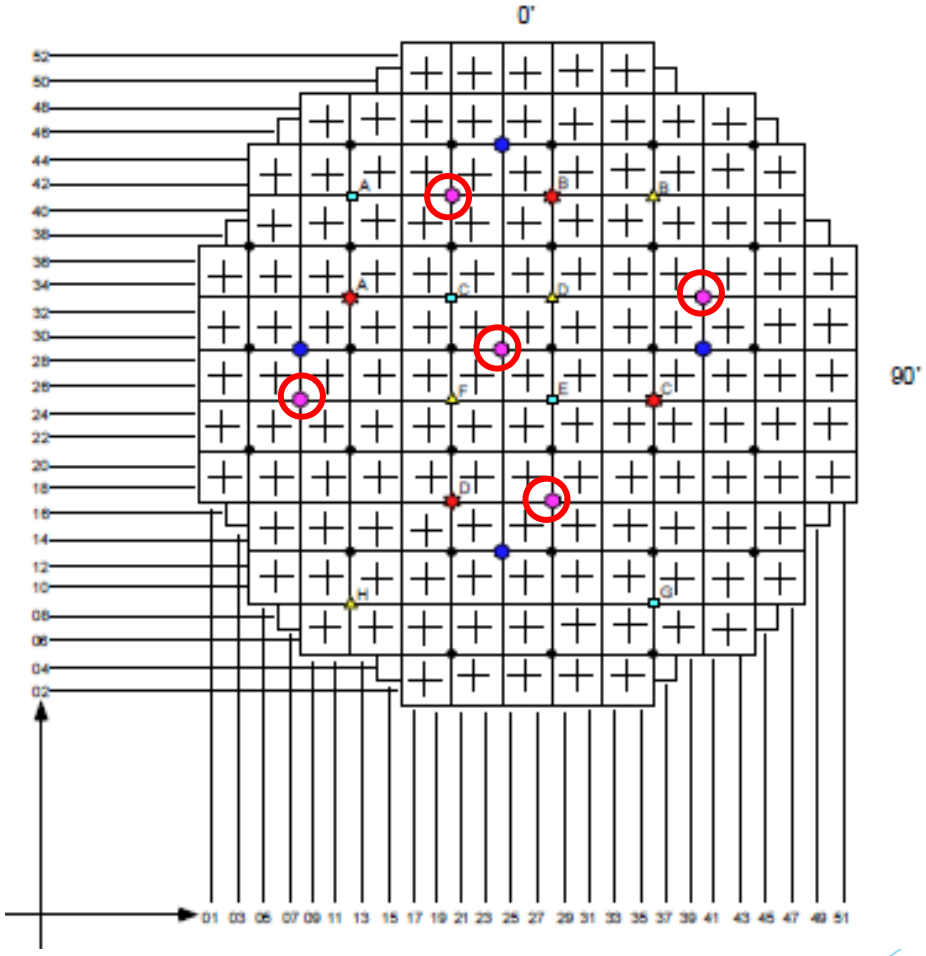
PWR neutron sources inserted into reactor core via control rod guide tubes

BWR neutron sources inserted into reactor core via in-core instrumentation guide tubes penetrating bottom of reactor vessel

Example Neutron Source Locations in PWR Core Fuel Assemblies



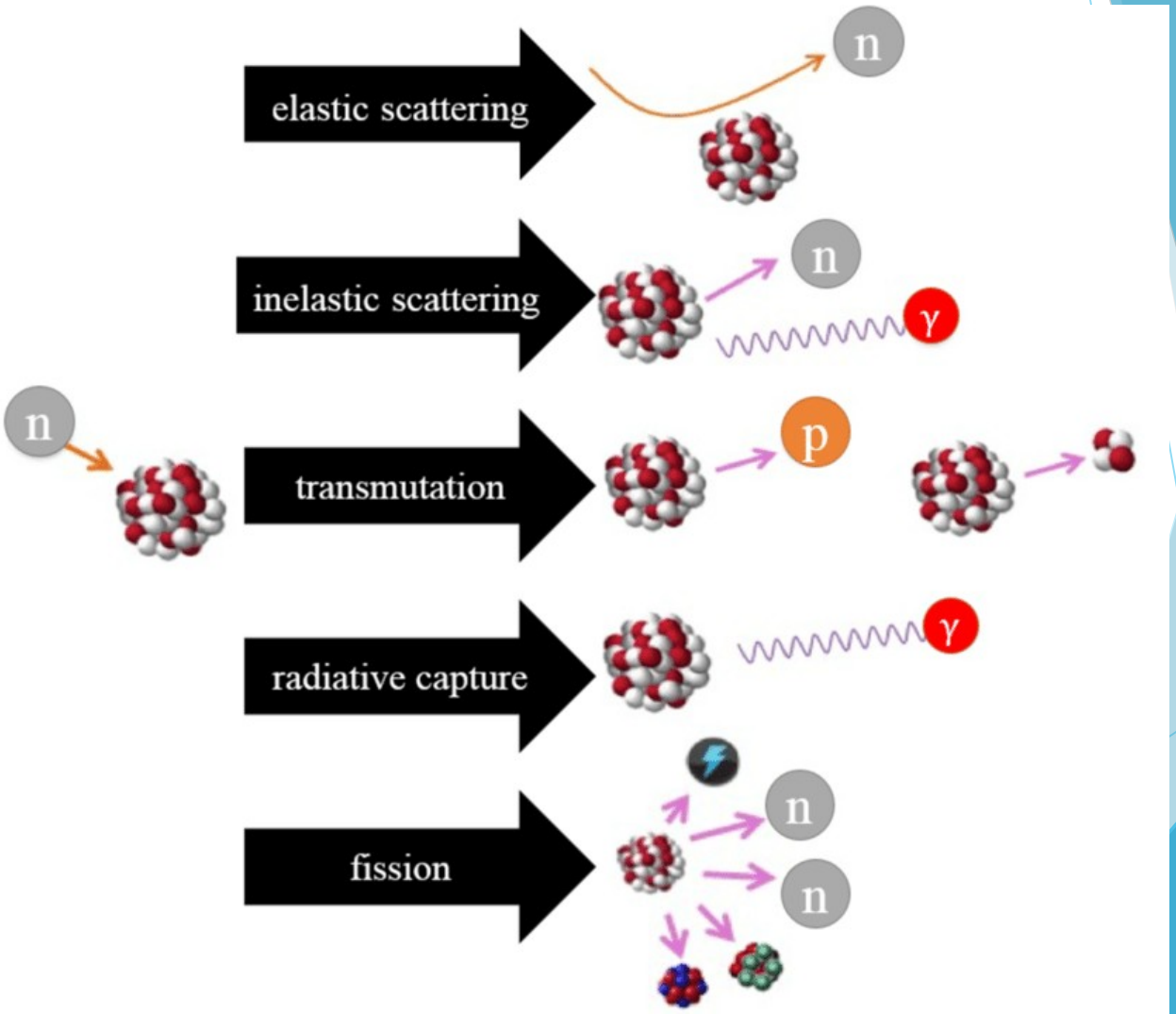
Example Neutron Source Locations in BWR Core Fuel Assemblies



LPRM – low power range monitor
IRM – intermediate range monitor
SRM – source range monitor

Neutron Interactions

Neutron Interactions with U-235



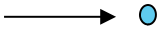
Neutron Interaction Cross-sections

- Neutron-nucleus collision may result in variety of reactions (e.g., scattering, capture, etc.)
- Concept of nuclear interaction *cross-section* – effective cross-sectional area of spherical nucleus should be proportional to probability of interaction with incident neutron
- Every nuclide has measured cross-section for each possible type of interaction and each incident neutron energy
  - Cross-sections vary significantly from nuclide to nuclide, and drastically with neutron energy

Illustration of Relative Nucleus Cross-sectional Area vs. Incident Neutron Energy

High Energy Neutron

Relative Cross-sectional Area of Nucleus for Nuclide X

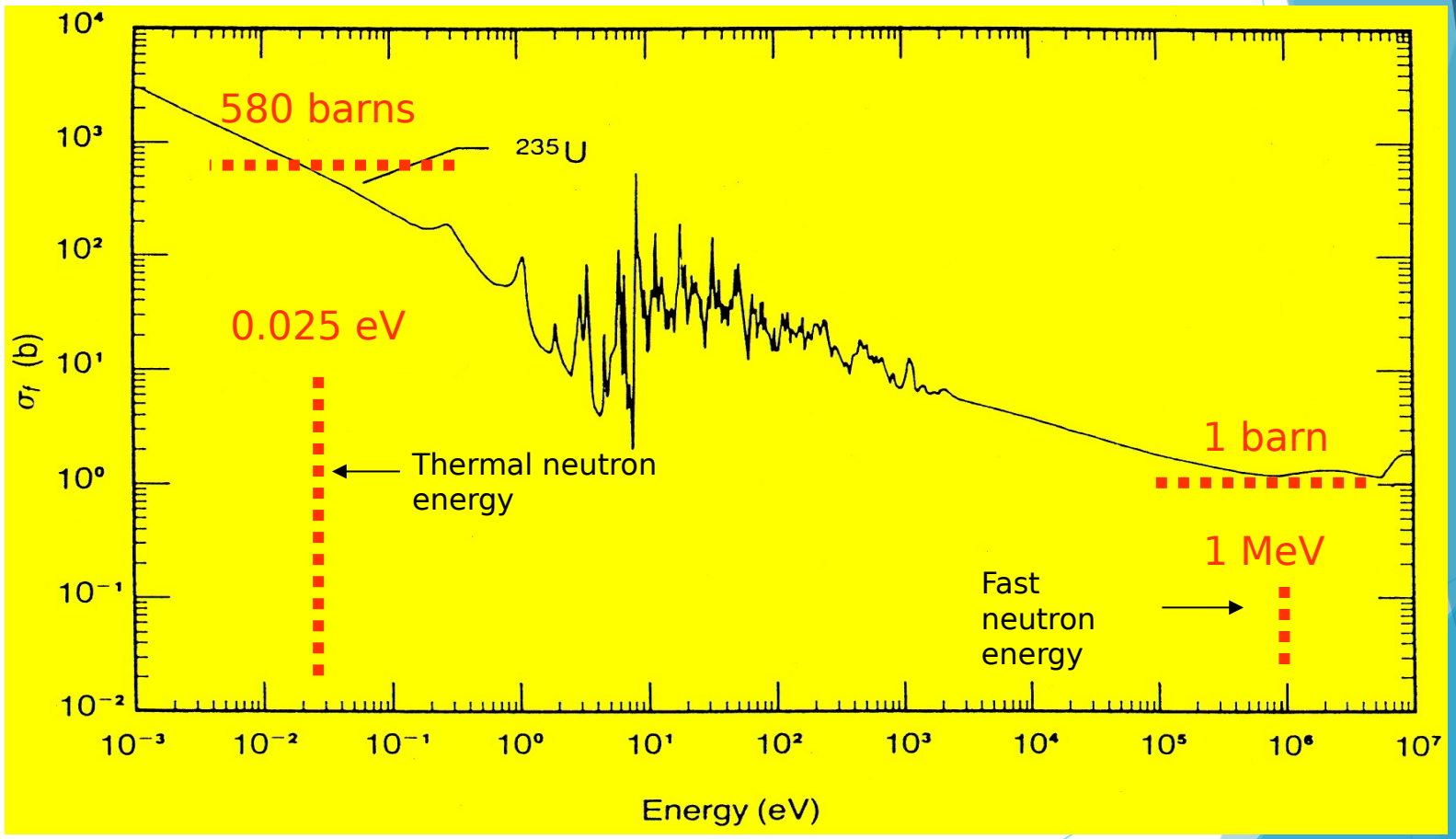


Low Energy Neutron

Neutron Interaction Cross-sections (cont.)

- Typical nuclide reaction cross-sections have values on order of 10^{-24} cm^2 (defined as 1 barn)
  - Barn is approximate cross-sectional area of U-235 nucleus for fission neutrons with energy of 1 million electron volts (next slide)
- Interaction cross-sections are function of incident neutron energy
  - Dependency can be fairly simple or extremely complex as illustrated in next slide graphic of fission cross-section for U-235 vs. neutron energy

Uranium-235 Fission Cross-section vs. Neutron Energy



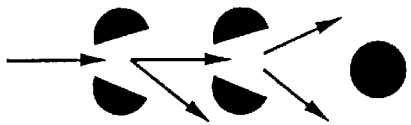
Controlling Heat Generation in Nuclear Fuel

Since nuclear reactor contains sufficient nuclear fuel for up to two years of continuous operation, rate at which thermal energy generated must be carefully managed by controlling fission reaction rate (i.e., neutron population) in fuel

- Fission reaction rate characterized by *neutron multiplication factor*, k (ratio of neutrons produced by fission in one generation to number of neutrons lost through absorption and leakage in preceding generation); cannot be measured directly
- Reactor operation with self-sustaining chain reaction (k = 1) defined as critical reactor condition

Subcritical

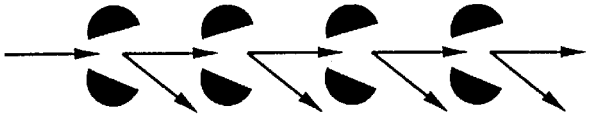
k < 1



Too much neutron absorption; no sustained chain reaction

Critical

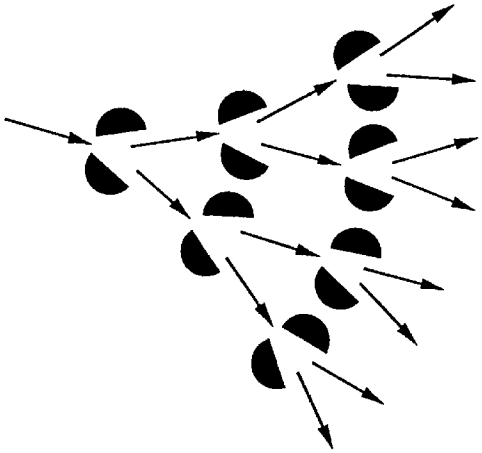
k = 1



Constant neutron population; sustained chain reaction

Super-critical

k > 1



Too little neutron absorption; exponentially increasing neutron population

Power Reactor Neutron Balance

- Neutron population in nuclear reactor at any time described by equation

Rate of neutron population increase	=	Rate of neutron production	–	Rate of neutron absorption	–	Rate of neutron leakage

or Accumulation = Production – Losses

- Neutron multiplication factor defined as

$$k = \frac{\text{neutron production}}{\text{neutron losses}}$$

- Just critical condition (k = 1) achieved at any fission rate (i.e., power level) as long as neutron production exactly balanced by neutron losses

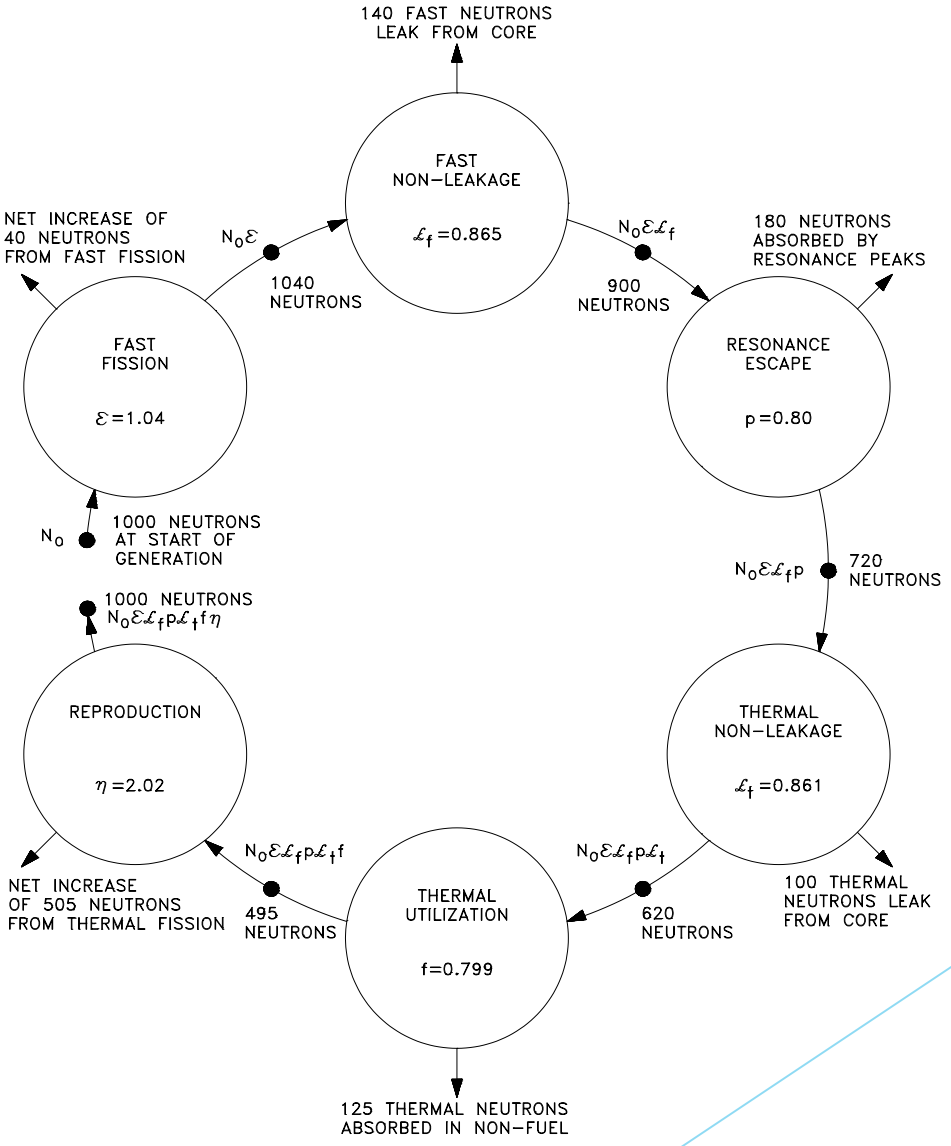
Power Reactor Neutron Balance (cont.)

- Power reactors require provisions for adjusting neutron balance
  - Must be critical (k=1) to produce steady-state power
  - Must become supercritical (k>1) to increase power
  - Must become subcritical (k<1) to reduce power and/or be shut down
  - Must be able to compensate for fuel depletion over time
- Understanding neutron life-cycle critical to neutron balance considerations in reactor design and operation

Neutron Life-cycle

Neutron Life-cycle with k = 1

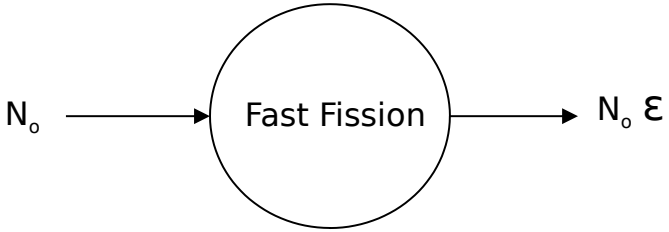
Start of Neutron Generation



Controlling Fission Reaction Rate

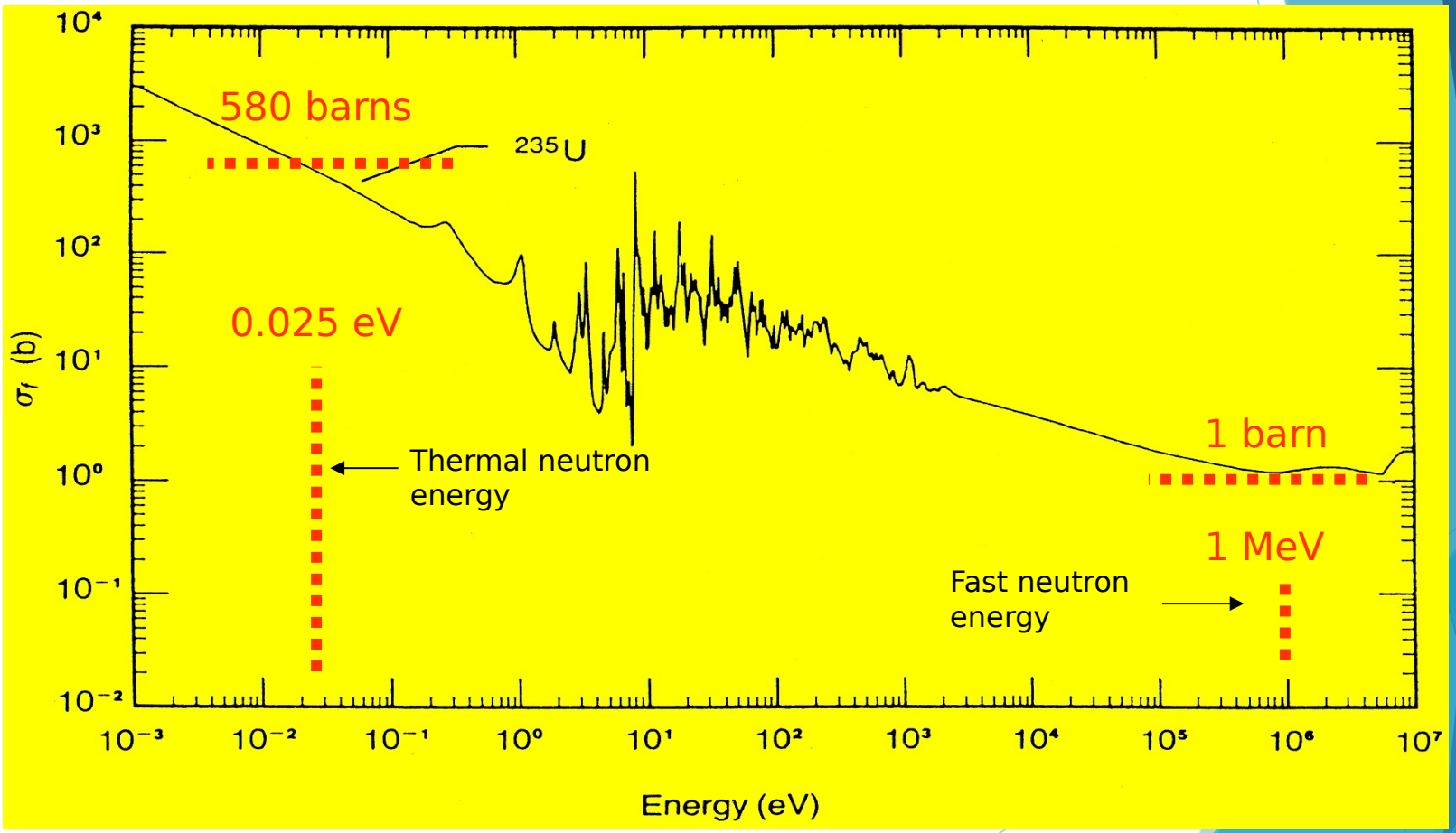
- When k not equal to exactly one, neutron population will change and cause change in reactor power level (i.e., energy generation rate)
- Value of system neutron multiplication factor can be determined from mathematical expression for neutron life-cycle events, $k = \varepsilon \mathcal{L}_f p \mathcal{L}_t f \eta$, where:
  - ε is fast fission factor
  - $\mathcal{L}_f$ is fast non-leakage probability
  - p is resonance escape probability
  - $\mathcal{L}_t$ is thermal non-leakage probability
  - f is thermal utilization factor
  - η is reproduction factor

Fast Fission Factor

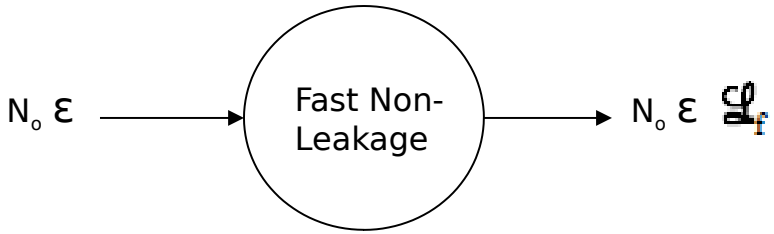


- Fast fission caused by neutrons in *fast* energy range – probability of U-235 capturing *fast* neutron (≈1 MeV) that produces fission is low compared to probability of capturing *thermal* neutron that produces fission (next slide graphic)
- Results in small increase in *fast* neutron population

Uranium-235 Fission Cross-section vs. Neutron Energy

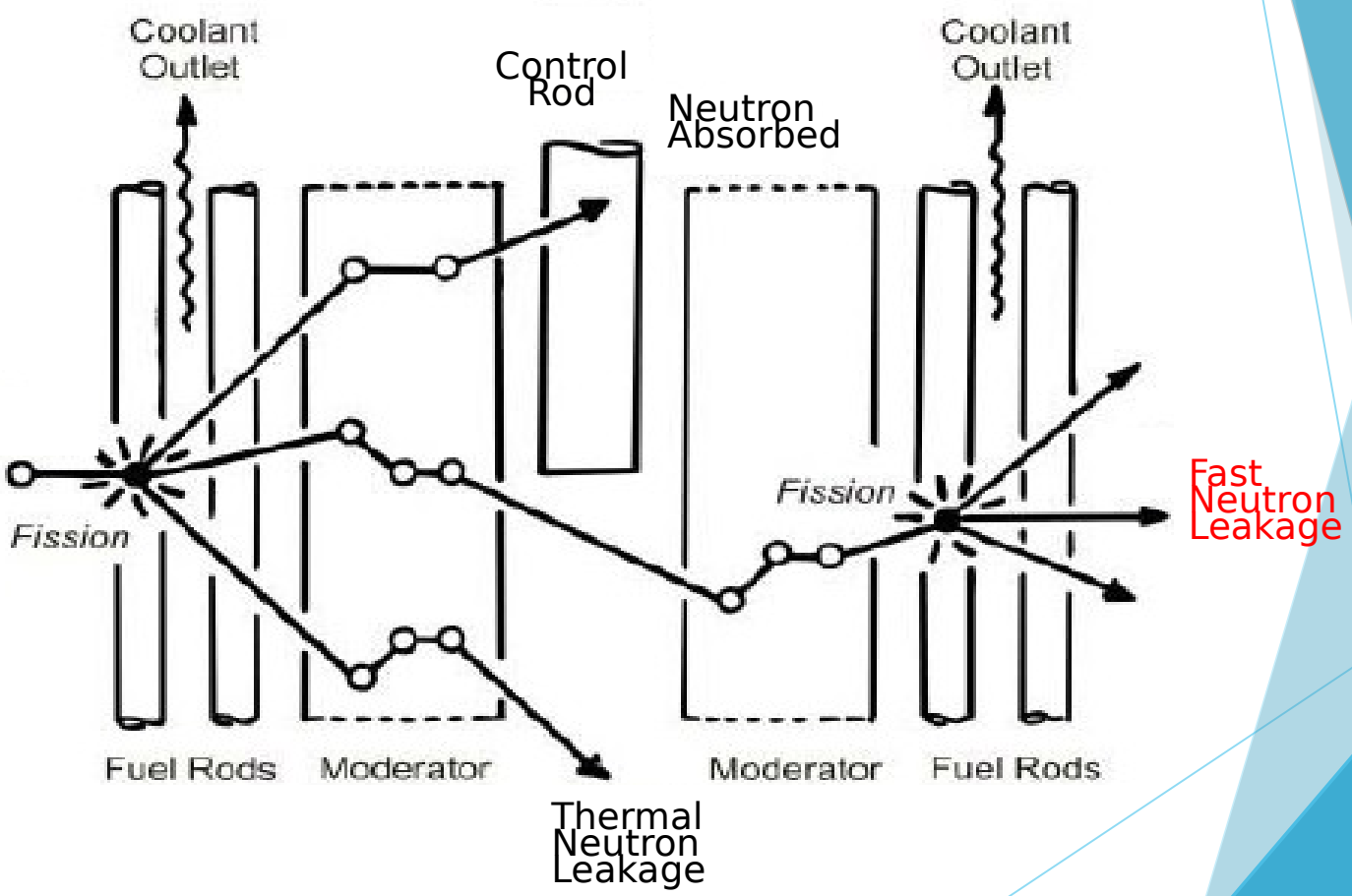


Fast Non-Leakage Probability

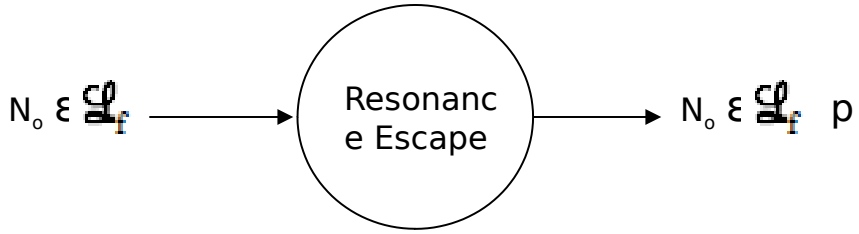


- Some *fast* neutrons exit (i.e., “leak” from) reactor core before beginning energy loss (or “slowing down”) process
- Results in decrease in *fast* neutron population

Fast Neutron Leakage Illustration

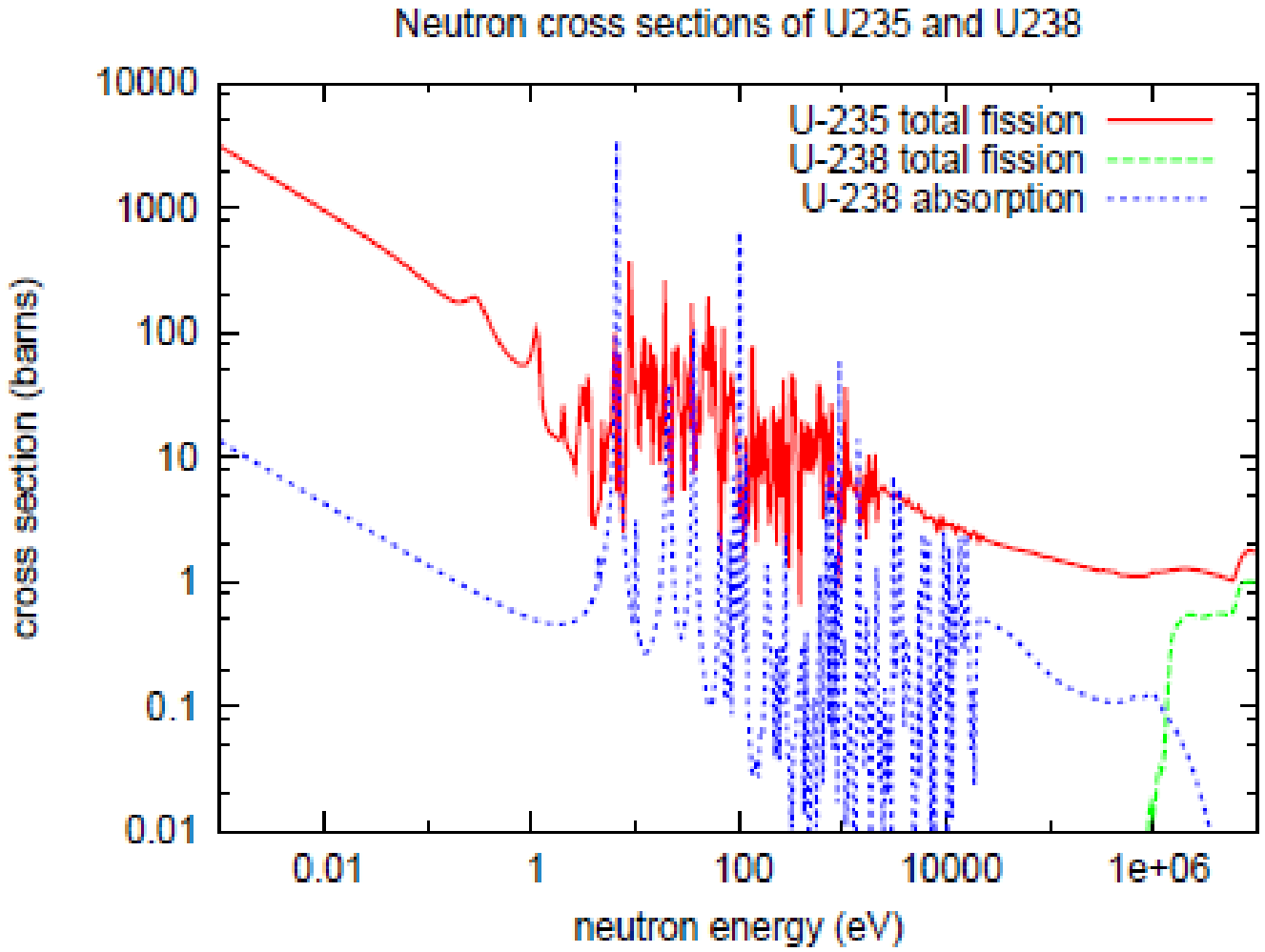


Resonance Escape Probability

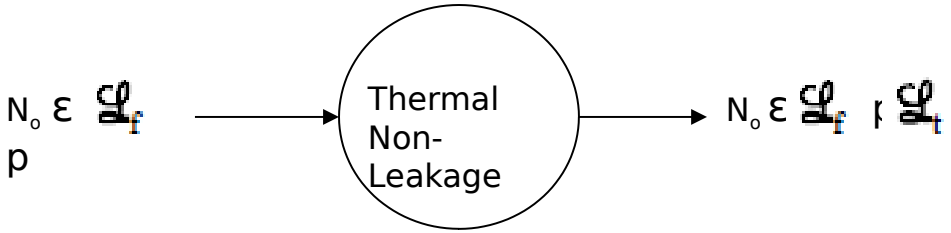


- While slowing down in energy region of U-238 absorption resonance peaks (next slide graphic) some neutrons absorbed by U-238, which makes up greater than 95% of uranium in light-water reactor fuel
- Results in decrease in neutron population achieving *thermal* energy

U-238 Resonance Capture Illustration

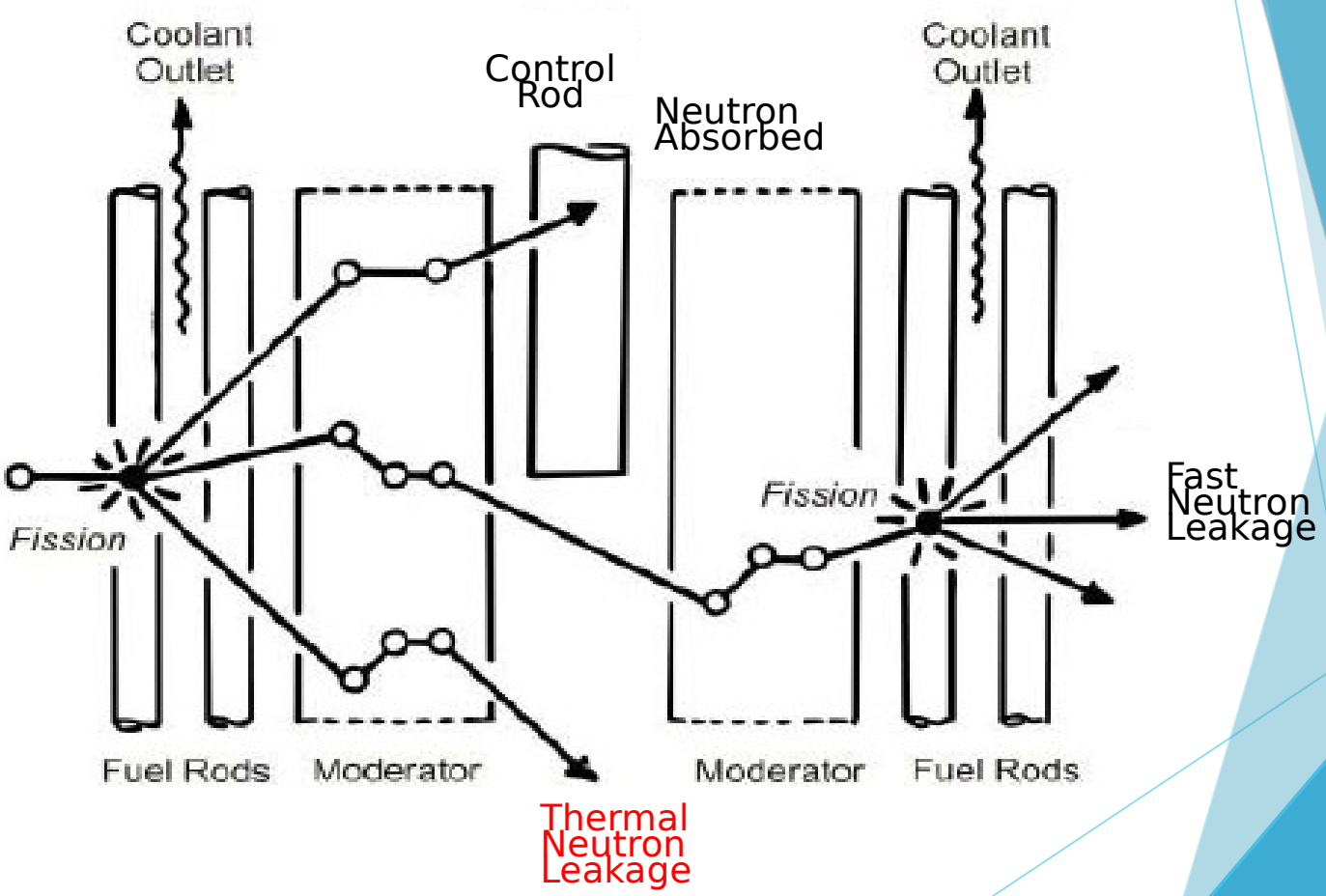


Thermal Non-Leakage Probability

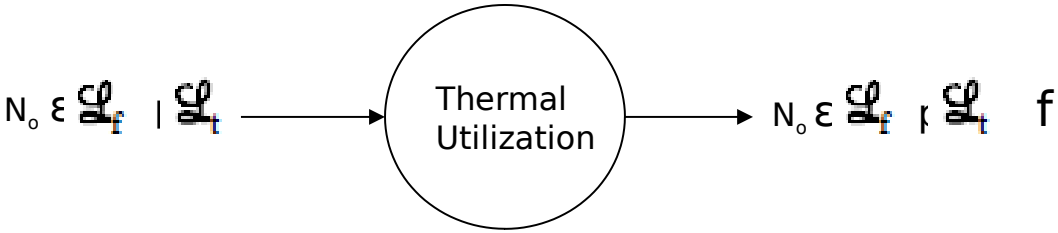


- Some neutrons exit (i.e., “leak” from) reactor core after reaching *thermal* energy
- Results in decrease in *thermal* neutron population

Thermal Neutron Leakage Illustration

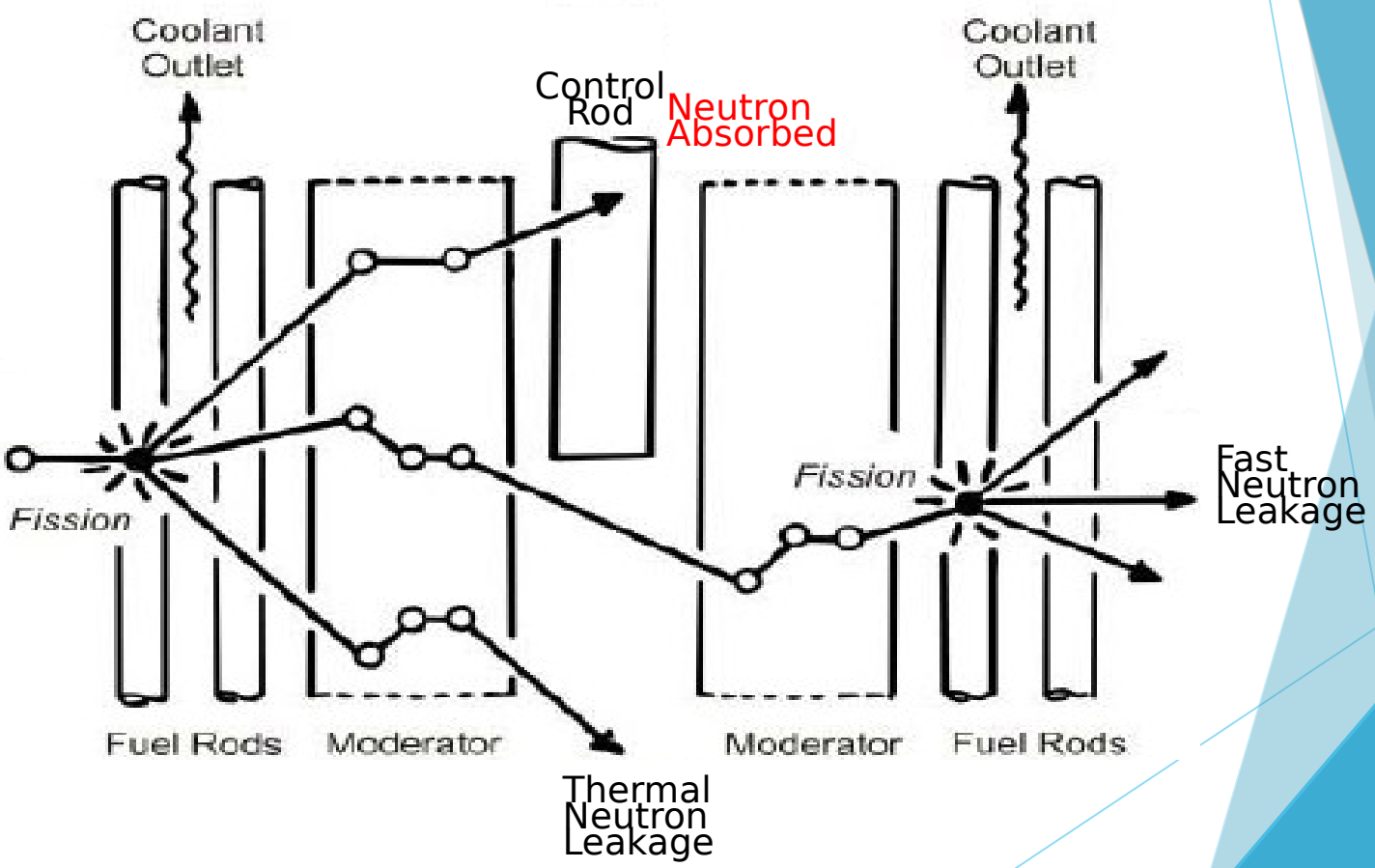


Thermal Utilization Factor



- Some neutrons absorbed in reactor core materials
  - All reactor materials have some propensity for thermal neutron capture
  - Movable control rods provide dynamic regulation of neutron population (reactor power level)
- Results in decrease in *thermal* neutron population

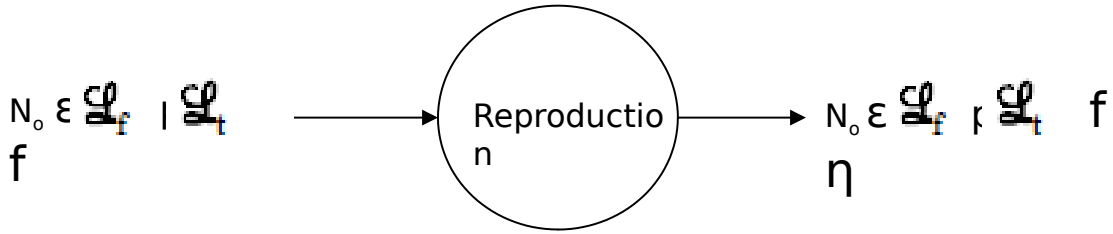
Thermal Neutron Absorption Illustration



Cross-sections of Reactor Materials

Material		Cross-sections in barns			
		Thermal neutrons (E = 0.025 eV)		Fast neutrons (E > 100 eV)	
		Fission	Capture	Fission	Capture
Fissile	U-235	585.086	98.6864	1.9041	0.55549
	Pu-239	747.401	270.329	1.7973	0.49614
Fertile	U-238	0.00001177	2.71692	0.042758	0.33188
Clad	Zirconium		0.185396		0.0265766
	Steel		3.08668		0.0170228
Coolant	Light Water		0.664		0.00051554
	Heavy Water		0.0013		0.00011459
Control Rod	B-10		3840.0		2.73462
	Cadmium		2524.15		0.266766
Fission Products	Xe-135		2636300.0		0.0059985
	Kr-83		207.667		0.235944
	Sm-149		40144.3		1.91883

Reproduction Factor



- Most neutrons absorbed in nuclear fuel cause fission, but not all (proportional to ratio of fission cross-section to sum of capture and fission cross-sections)
- Results in next generation of *fast* neutrons
  - If larger than previous generation, k > 1
  - If smaller than previous generation, k < 1
  - If same as previous generation, k = 1

U-235 Cross-sections for Capture and Fission Interactions

	Thermal Cross-section (barn) E = 0.025 eV			Fast Cross-section (barn) E > 100 eV		
	Scattering	**Capture**	**Fission**	**Scattering**	**Capture**	**Fission**
U-235	<10	98.7	585	<4	0.56	1.9

Total absorption cross-section = capture cross-section + fission cross-section

Neutron Life-cycle Time

- Average (or mean) neutron generation time is time required for neutrons from one generation to cause fissions that produce next generation of neutrons
  - Fast neutrons slow to thermal energies or leak from system in $\approx 10^{-7}$ second
  - Thermal neutrons exist for $\approx 10^{-4}$ second before leaking from system or being absorbed
  - *Mean neutron generation time* for thermal reactors is then $\approx 10^{-4}$ second

Neutron Life-cycle Time (cont.)

- U-235 fission produces average of 2.42 neutrons per fission event
  - Majority are *prompt* neutrons emitted at time of fission
  - Small fraction are *delayed* neutrons, appearing up to a minute later
- Delayed neutrons have significant effect on thermal reactor *mean neutron generation time* – critically important to control of neutron population changes (reactor power changes)

Delayed Neutrons

Delayed Neutrons

Two types of neutrons released in fission

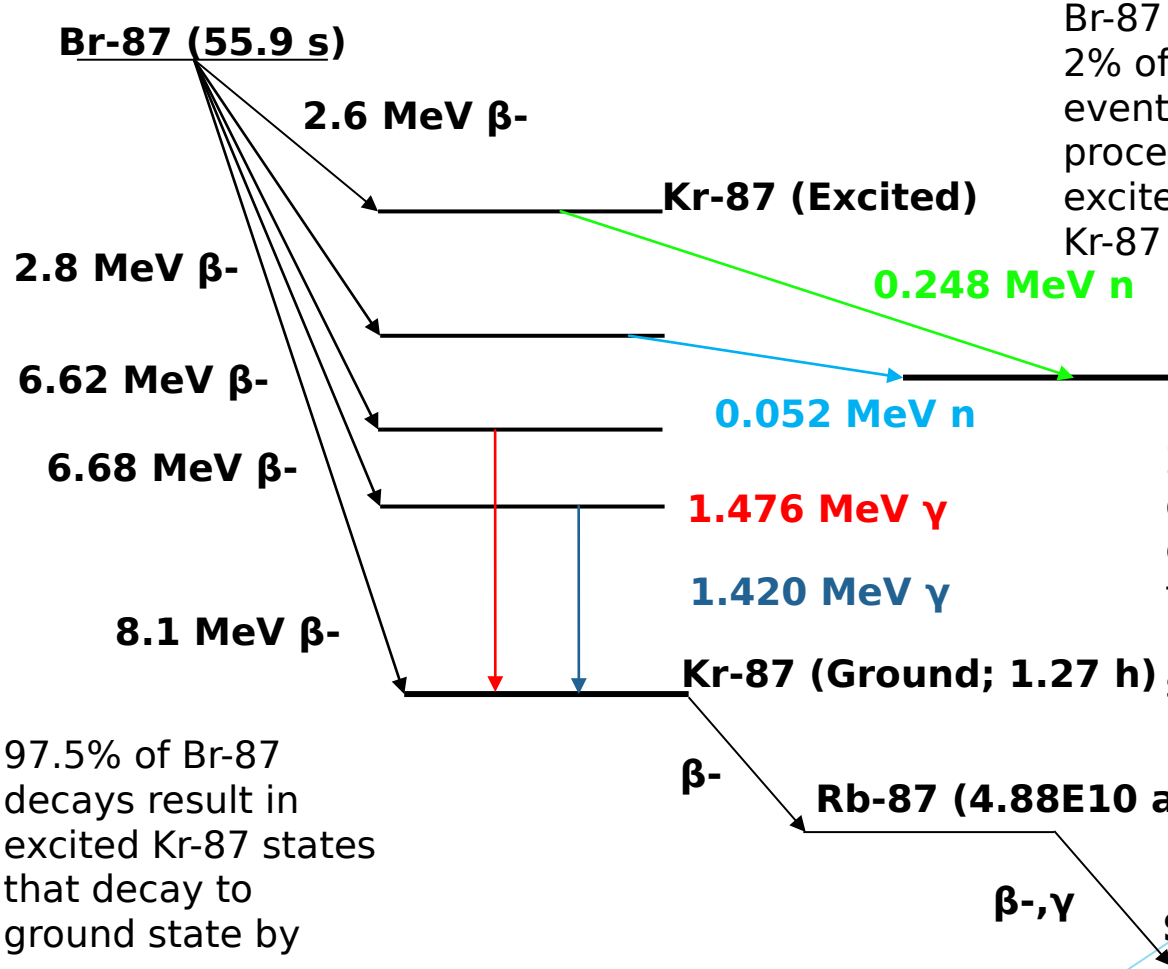
- Prompt neutrons
  - Greater than 99% of total fission neutrons
  - Released within 10^{-14} second of fission event
- Delayed neutrons
  - Expelled from daughter nucleus following certain fission fragment beta decays
  - Small but very significant contributor to time-dependent behavior of neutron population (decrease rate of change of reactor power over that which would occur if only prompt neutrons produced by fission)
  - Make power reactor control possible

Delayed Neutrons (cont.)

Half-life of fission fragment beta decay establishes delayed neutron generation time

- For example, Br-87 fission fragment is delayed neutron precursor with half-life of ≈56 seconds
- Br-87 decays by beta emission to multiple energy states of Kr-87 (next slide graphic)
- Two excited energy states of Kr-87 decay by neutron emission to stable Kr-86

Delayed Neutron Illustration



Br-87 formed in about 2% of U-235 fission events; its beta decay proceeds to multiple excited energy states of Kr-87

2.5% of Br-87 beta decays result in excited Kr-87 states that decay by neutron emission to stable Kr-86

97.5% of Br-87 decays result in excited Kr-87 states that decay to ground state by gamma-ray emission.

Delayed Neutron Fraction

- Single most important parameter in reactor kinetics for sustained, controlled fission chain reaction
- Total *delayed neutron fraction* (β) is 0.0065 for U-235 fission; prompt neutron fraction (1-β) is then 0.9935
- Delayed neutrons significantly increase neutron life-cycle time
  - Prompt neutron *lifetime* (i.e., thermal neutron lifetime) is $\approx 10^{-4}$ second
  - Average delayed neutron generation time is ≈12.5 seconds
  - Effective neutron generation time is weighted average;

    $\text{Time}_{avg} = \text{Time}_{p}\ (1-\beta) + \text{Time}_{d}\ (\beta) = 0.081$ second

System Reactivity

System Reactivity

- Because k = 1 always implies reactor criticality, k – 1 is excess multiplication with respect to critical condition
- Fractional excess multiplication, represented by (k – 1)/k or Δk/k, defines system *reactivity* (ρ)
  - All power reactors have “built-in” excess reactivity in nuclear fuel to allow for multiyear operation
  - Operational criticality control (k = 1, ρ = 0) maintained by reactivity control systems
- Reactivity is abstract concept developed to quantify system’s neutron multiplication behavior
  - Cannot be measured directly
  - Variations in system physical parameters result in reactivity changes (e.g., control rod movement, moderator density)
  - Reactivity changes can be either positive or negative

Relationship Between k, ρ, and System Criticality

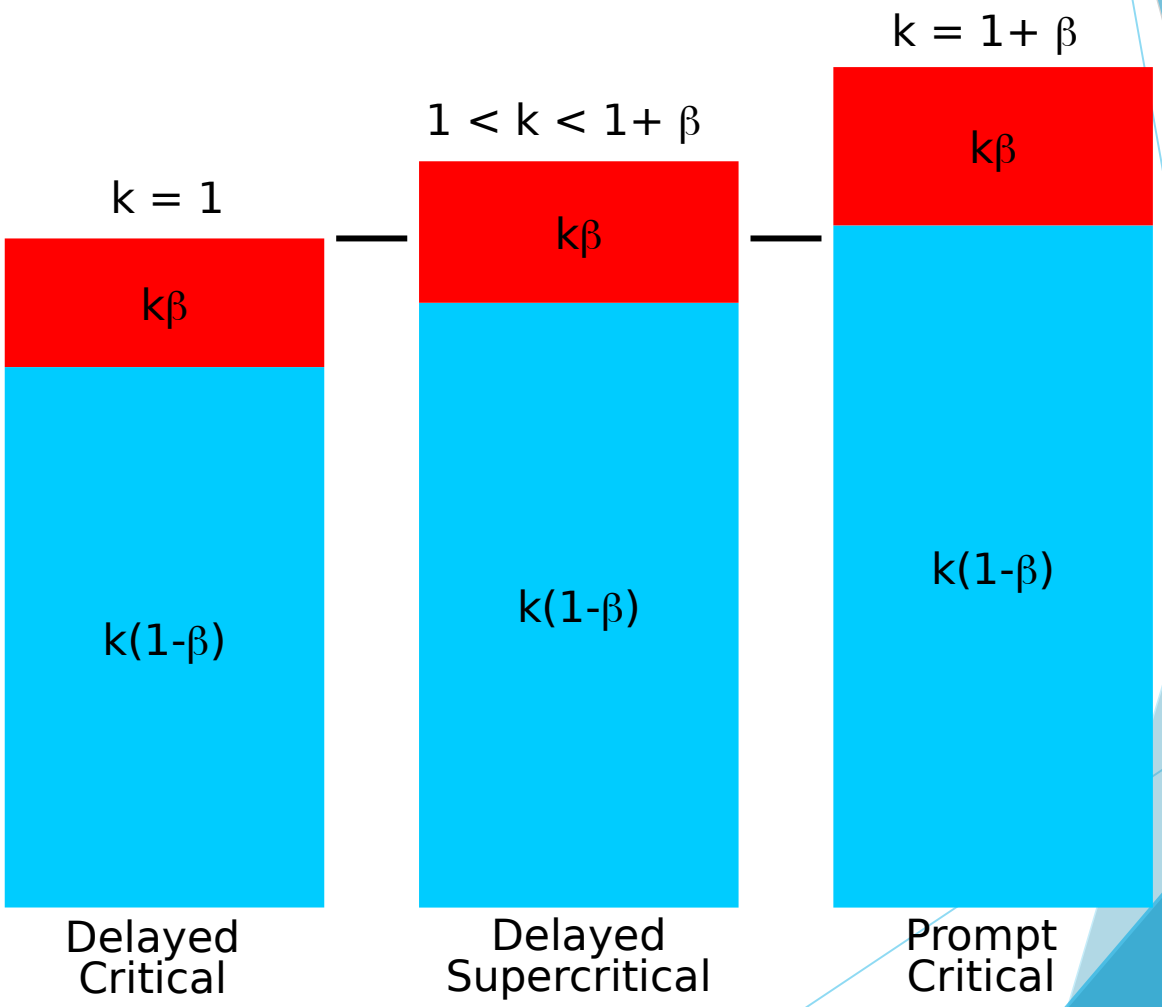
Multiplication Factor	Reactivity	State of Criticality
$k > 1$	$\rho > 0$	Supercritical
$k = 1$	$\rho = 0$	Critical
$k < 1$	$\rho < 0$	Subcritical

Reactor Criticality Achieved With Delayed Neutrons

Nuclear power reactors rely on delayed neutrons to hold fission reaction rate reasonably constant (i.e., maintain k ≈ 1)

- Reactors designed to ensure positive reactivity changes much smaller than delayed neutron fraction (0.65%)
  - Control rod reactivity worth and control rod drive speed both limited by design
- Ensures slow enough time-scale to permit control by human operator or mechanical system intervention
- Operating reactor fluctuates between being slightly *subcritical* and slightly *delayed-supercritical*, but must always remain below *prompt critical* (next slide graphic)

Role of Delayed and Prompt Neutrons in Critical Reactor Conditions



Practical System Criticality Control

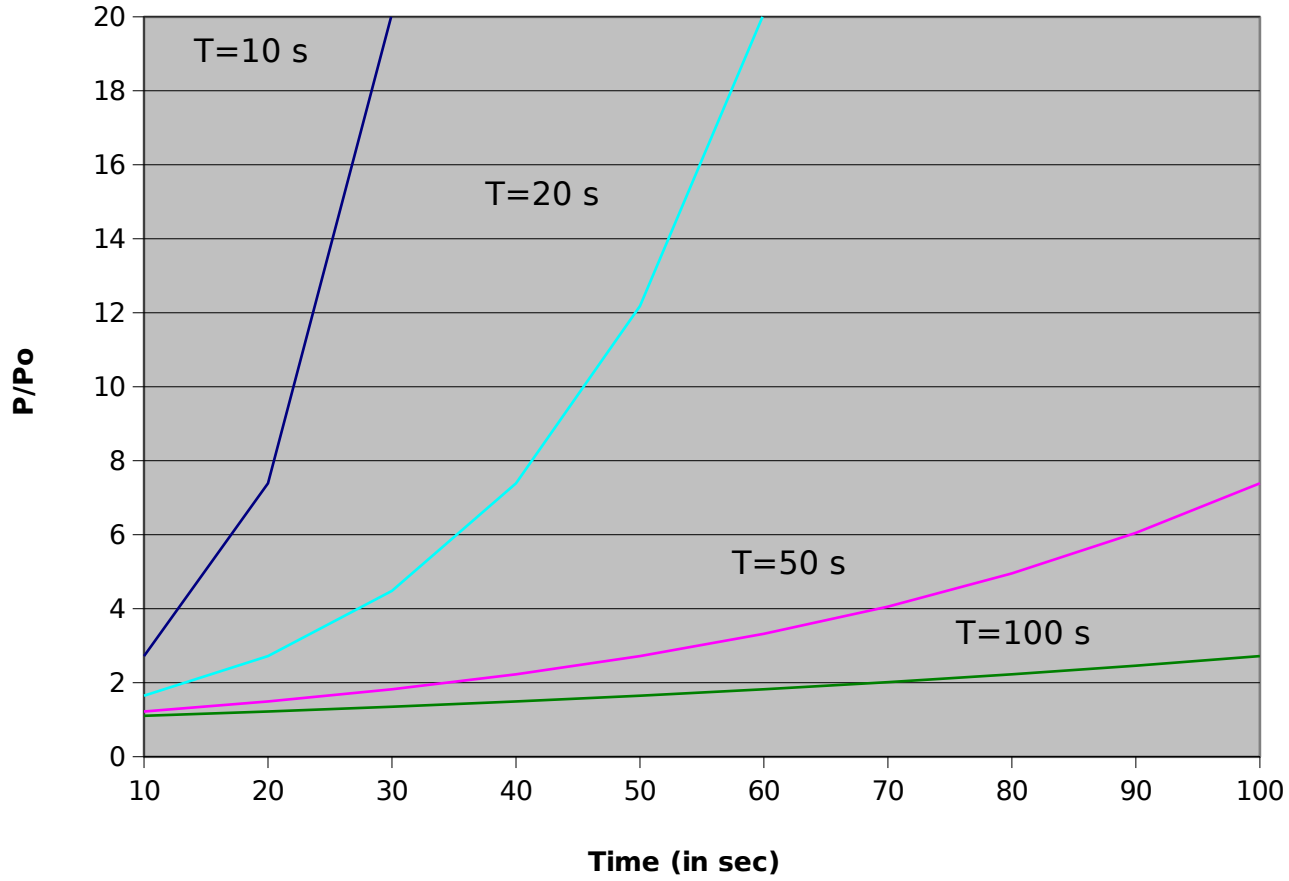
Reactor Period

- Since k and ρ can't be measured directly, reactor operator must rely upon measurable parameter called reactor period to
  - achieve stable (critical) operating condition
  - safely change power levels by changing system reactivity
    (e.g., by movement of control rods)
- Reactor power level changes in time-dependent manner according to equation $P = P_o e^{t/T}$, where
  P_o = initial reactor power
  t = time of reactor power transient in seconds
  T = reactor period in seconds
  - Rate of power increase inversely proportional to T

Reactor Period (cont.)

- *Reactor period* (T) is time required for reactor power (or neutron density) to change by factor of e (i.e., factor of about 2.7)
  - Measure of how rapidly reactor power level changes
    due to positive or negative reactivity changes in reactor core (next slide graphic)
- Period can be either positive (causes power increase) or negative (causes power decrease), and generally expressed in seconds

Power Increase vs. Time as a Function of Reactor Period (T)



Reactor Period (cont.)

- Inversely proportional to change in reactivity; i.e., the larger the positive reactivity change, the smaller the reactor period (next slide graphic), and the faster the increase in reactor power
- Determined in practice by measuring with a stopwatch time required for reactor power to double (i.e., doubling time) following a specified control rod withdrawal ($T = t_d/\ln 2$, where t_d is stopwatch-measured doubling time)

Estimated Reactor Period for Positive Reactivity Changes*

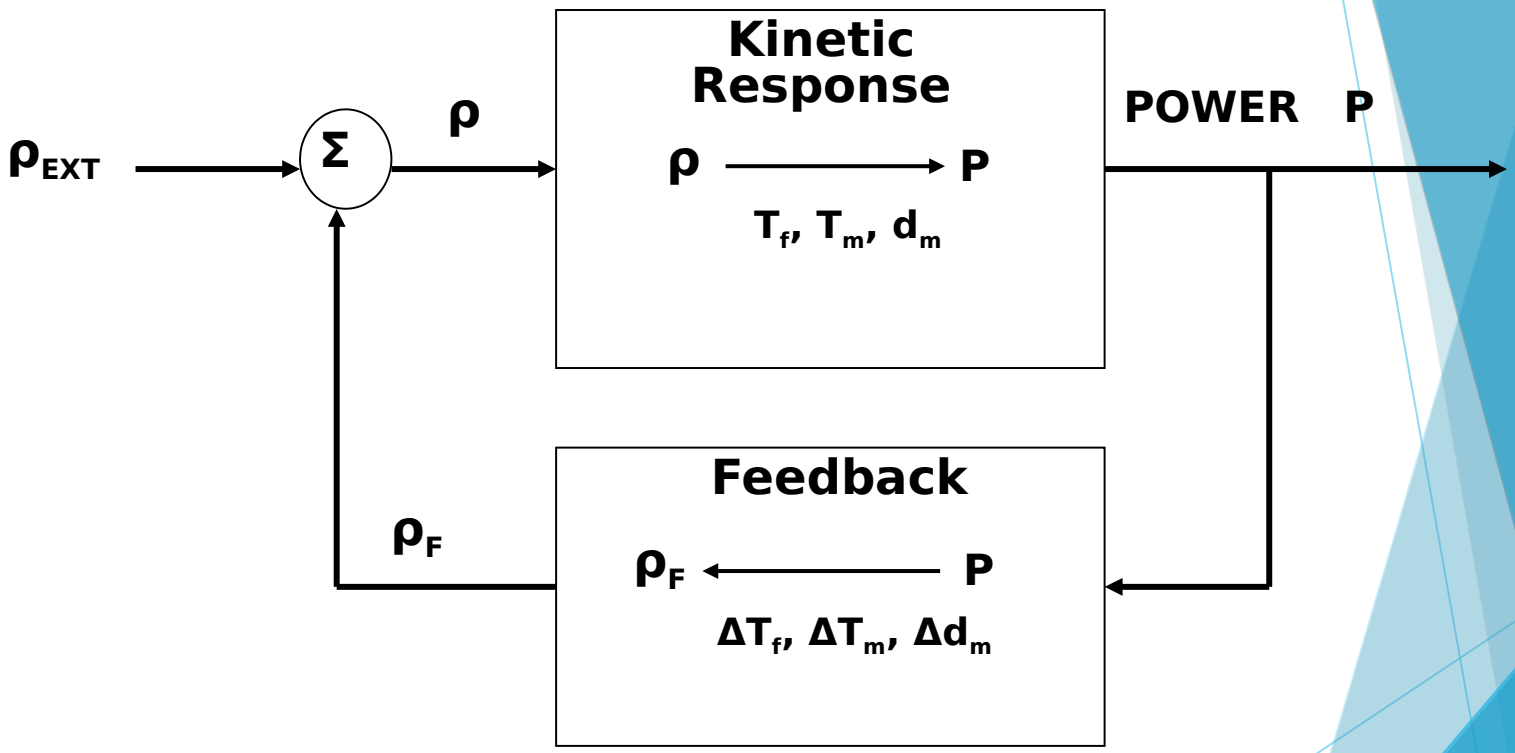
* Estimates using one delayed neutron group are for a typical light-water reactor (LWR) using slightly enriched uranium fuel

Practical Criticality Control Assured Through Proper System Design

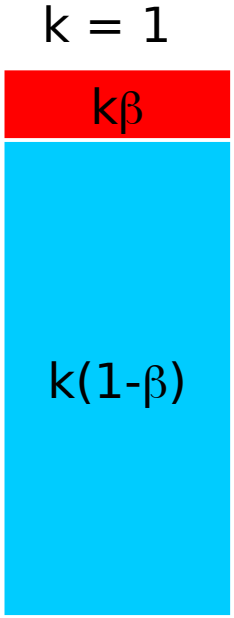
Proper integration of composition and geometry of nuclear fuel, reactor core materials, thermal-hydraulics, and reactivity control features in reactor design

- Assures that planned and inadvertent reactivity changes are small (<< 0.65%)
  - Individual control rod reactivity worth and control rod drive speed both limited
- Assures that reactivity feedback mechanisms are negative and operate on short time scales (e.g., fuel temperature, moderator temperature, moderator density)

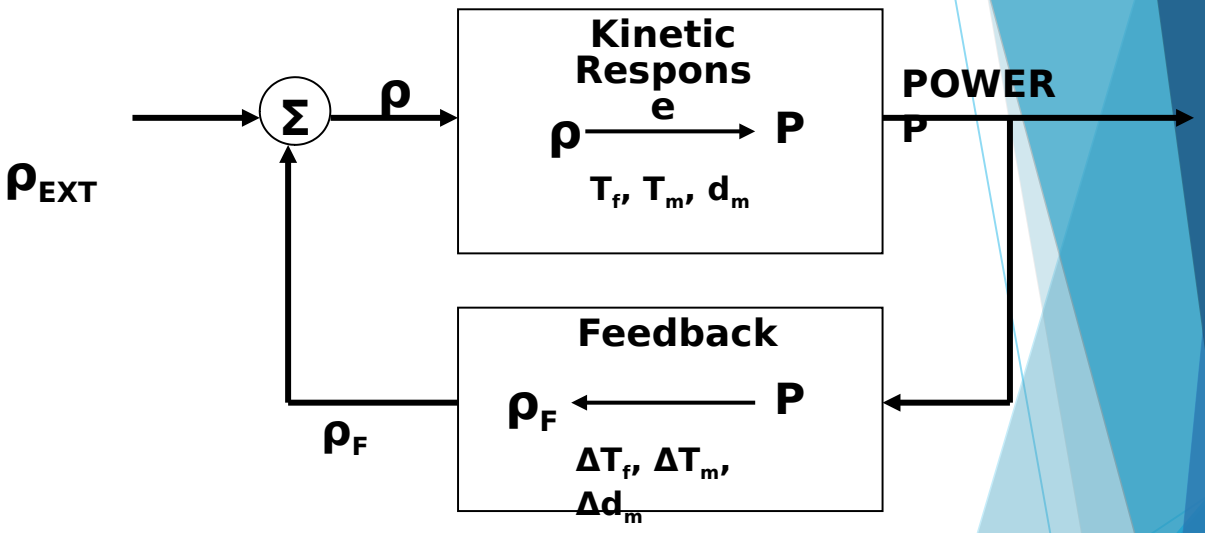
Reactivity Feedback Diagram



Steady-state Power Operation



Delayed Critical Condition



$ρ_{EXT} = 0, \quad ρ_F = 0$

Mathematically

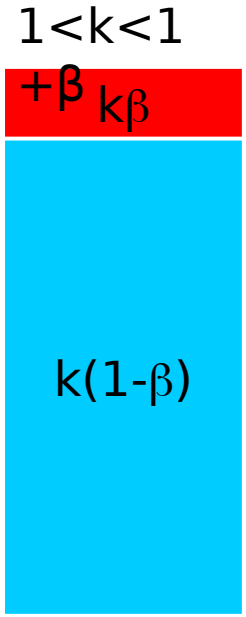
For steady-state operation, $P/P_0 = e^{t/T} = 1$

Since $e^0 = 1$, then $t/T = 0$

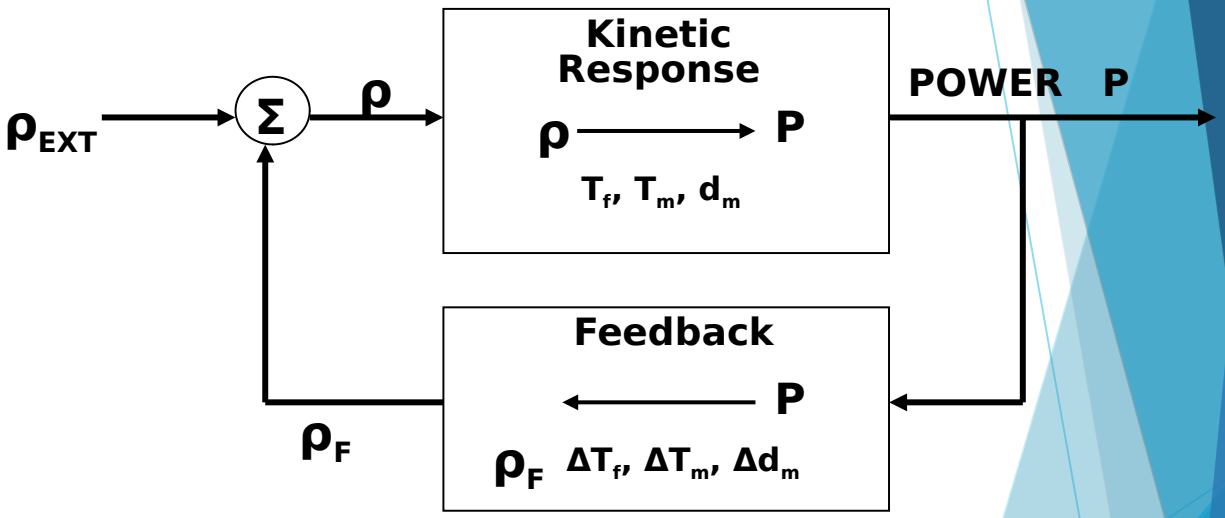
In steady-state operation, power level does not change as time marches on (i.e., $t \neq 0$)

Therefore, $t/T = 0$ implies Reactor Period (T) is infinite

Step Change Power Increase

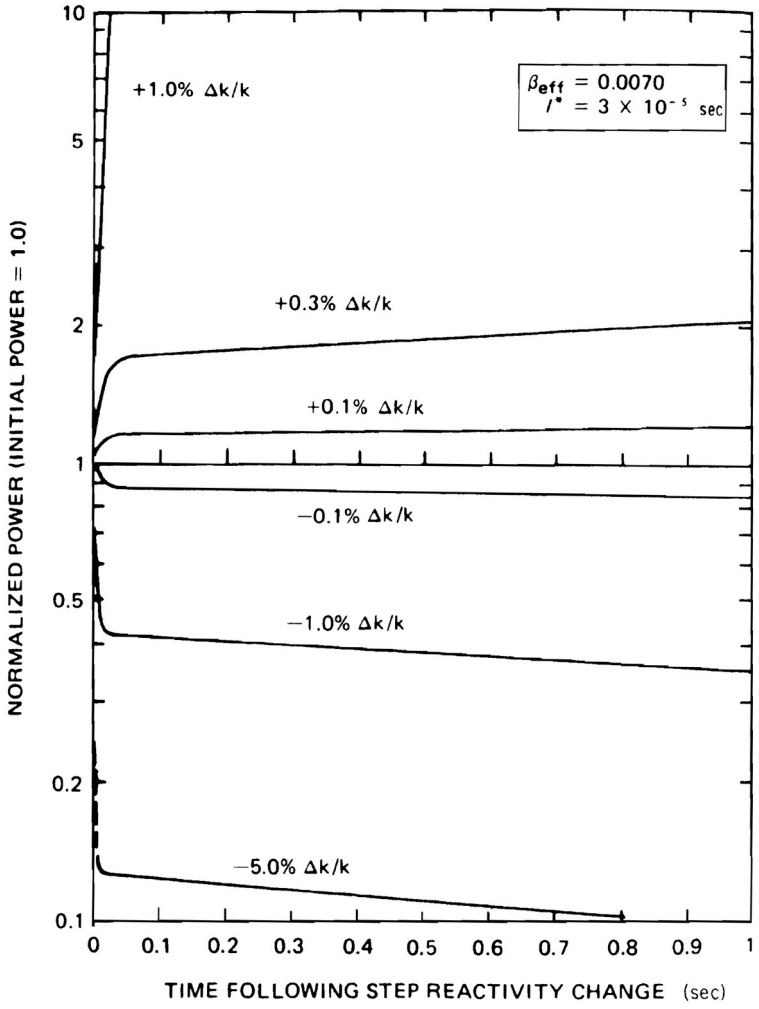


Delayed Supercritical Condition



0 < ρ_{EXT} << β, ρ_F < 0

Time-Dependent Power Behavior for Various Reactivity Changes



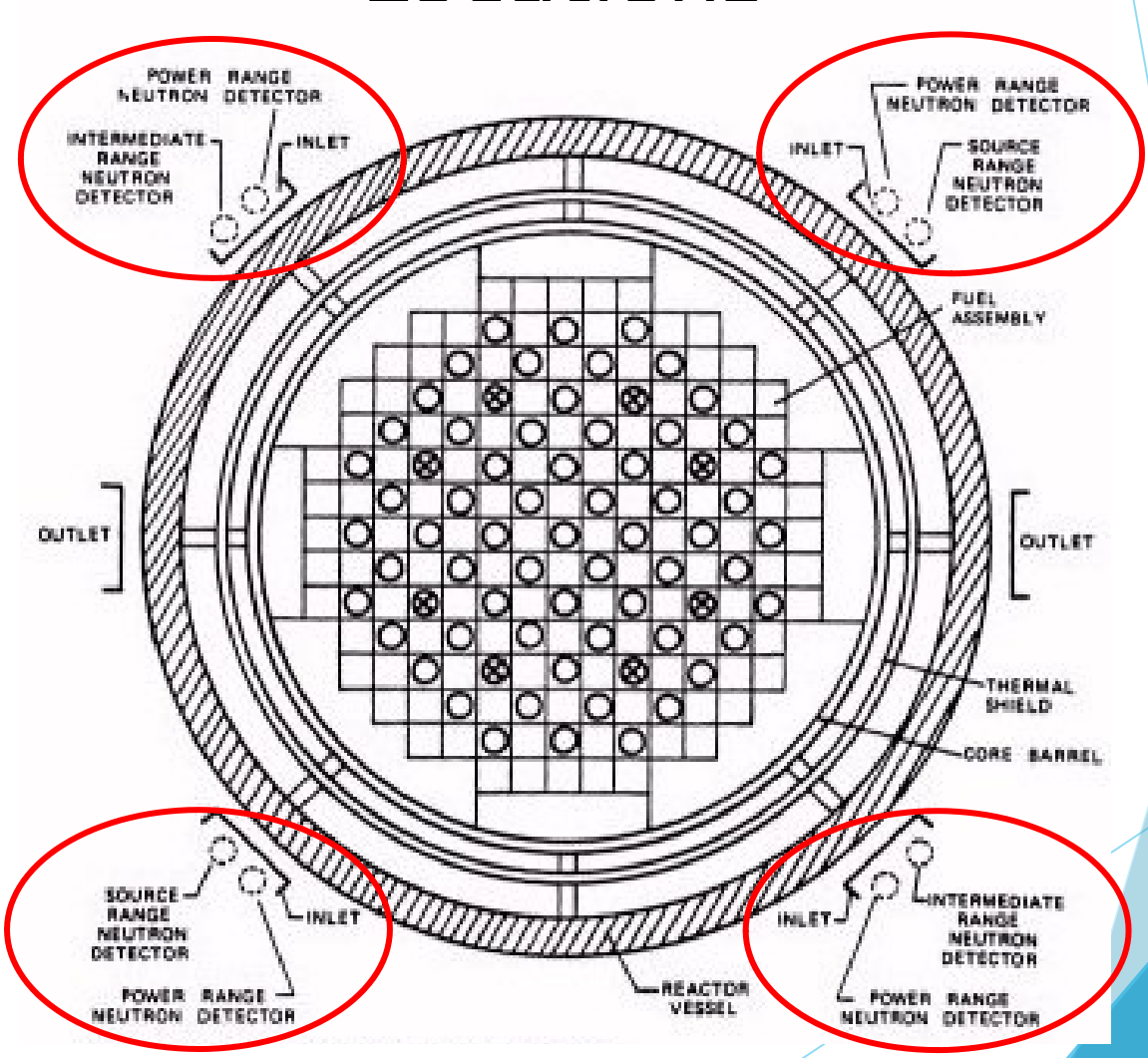
Monitoring Reactor Power and Reactor Control Instrumentation

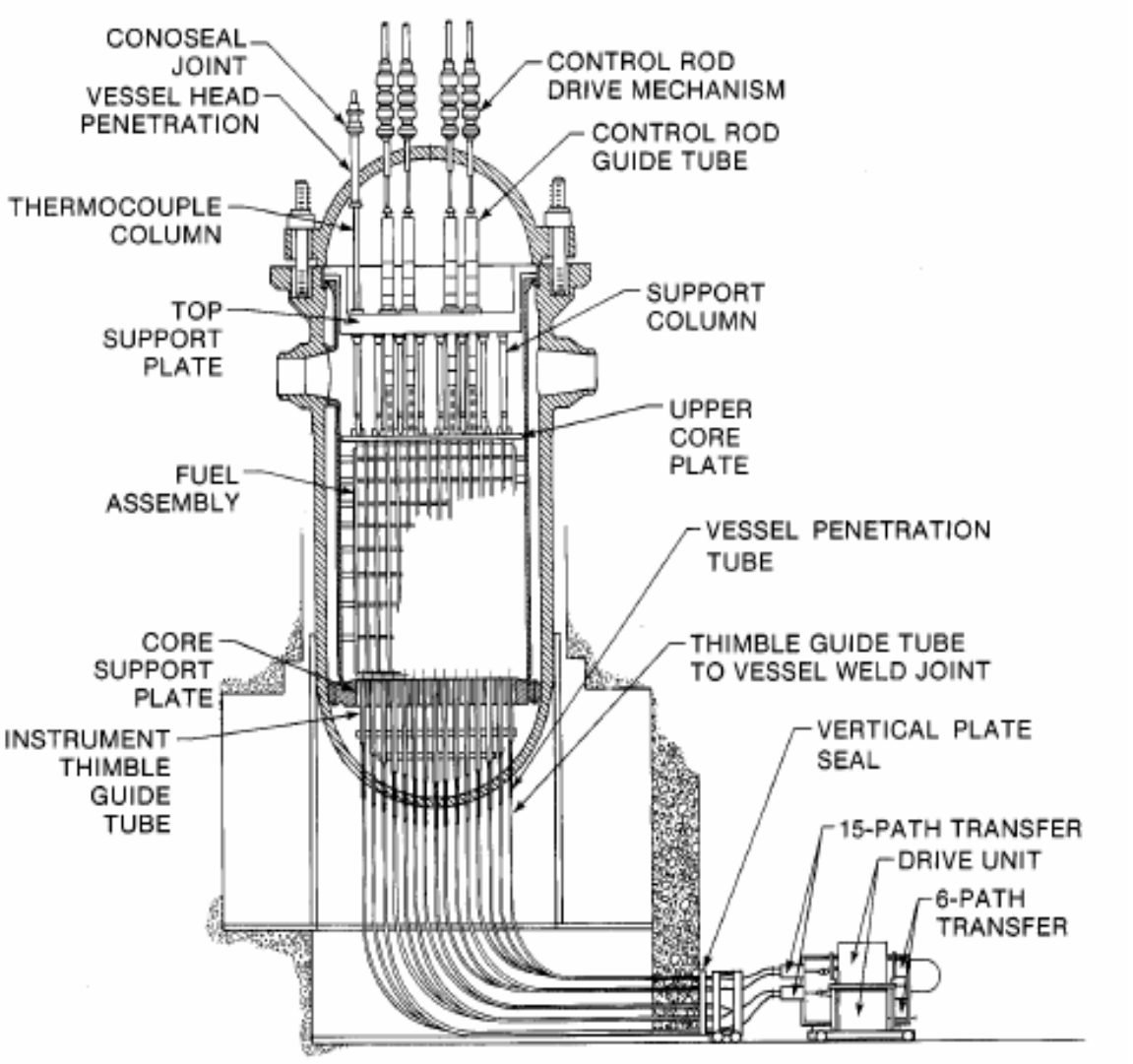
(How Neutron Population is Actually Monitored)

Neutron Detection Instrumentation

- Reactor operators, automatic reactor control system, and reactor safety mechanisms must have instantaneous information about reactor power level and rate of power level increases to ensure reactor heat generation rate does not exceed design capacity of reactor heat removal system
- Reactor power level (fission rate) at any instant and any core location proportional to neutron flux (i.e., number of neutron interactions with nuclei in 1 cm^2 of target material per second)
- Reactor power level typically monitored by measuring neutron flux with both in-core and ex-core instrumentation having almost instantaneous response (other means of determining reactor power too slow for safe operation to be possible)

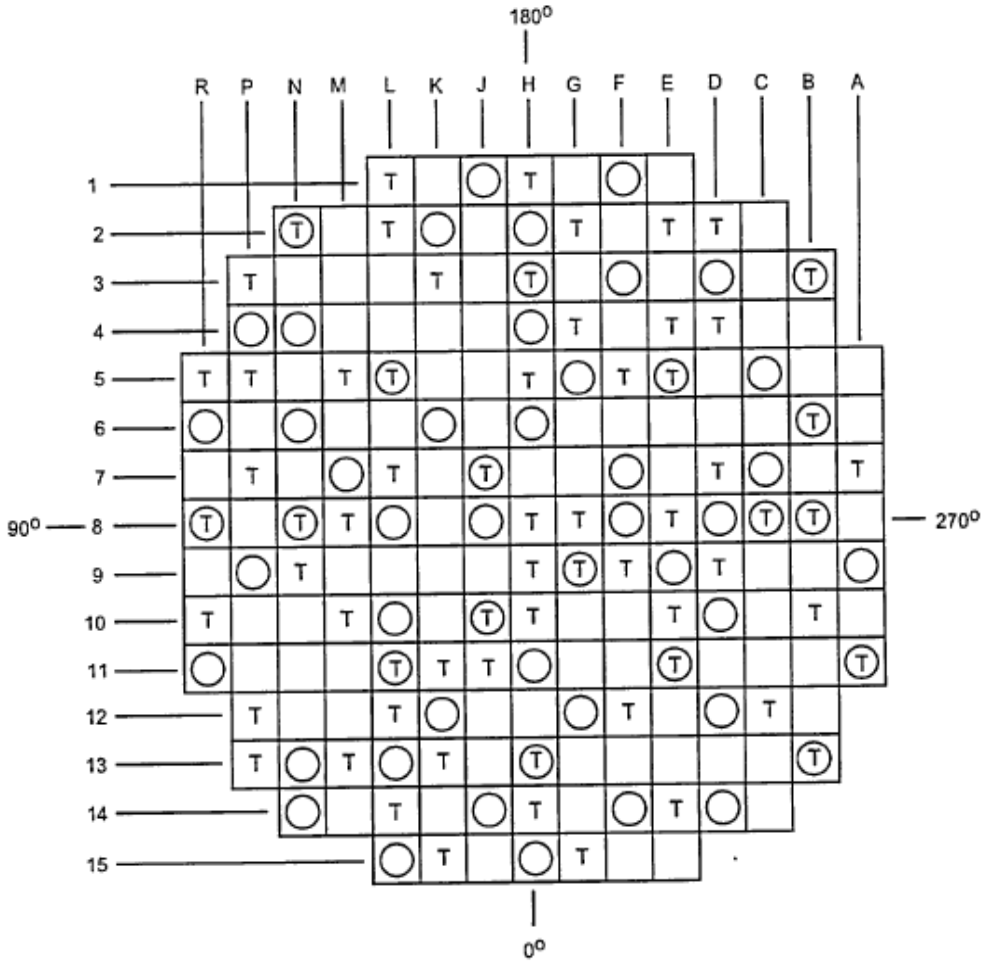
Typical Ex-core Neutron Detector Locations



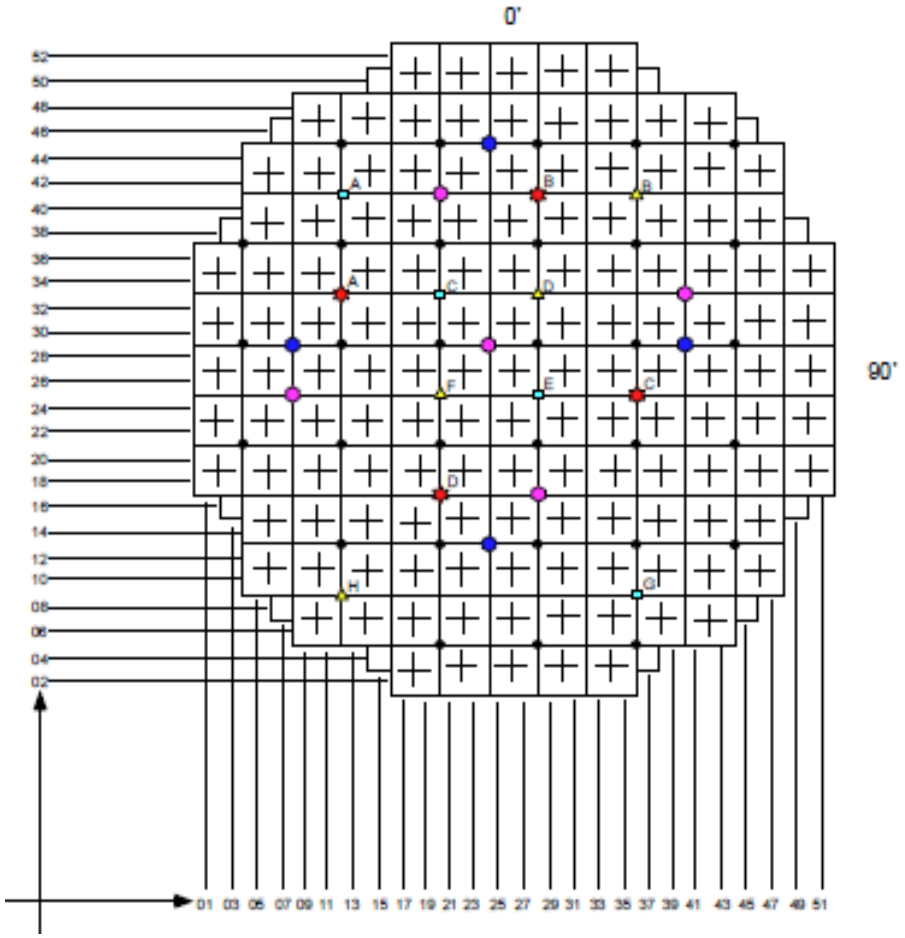


LWR in-core instrumentation inserted and removed from reactor core through instrumentation guide tubes that penetrate bottom of reactor vessel

Typical PWR In-core Neutron Detector Locations



Typical BWR In-core Neutron Detector Locations



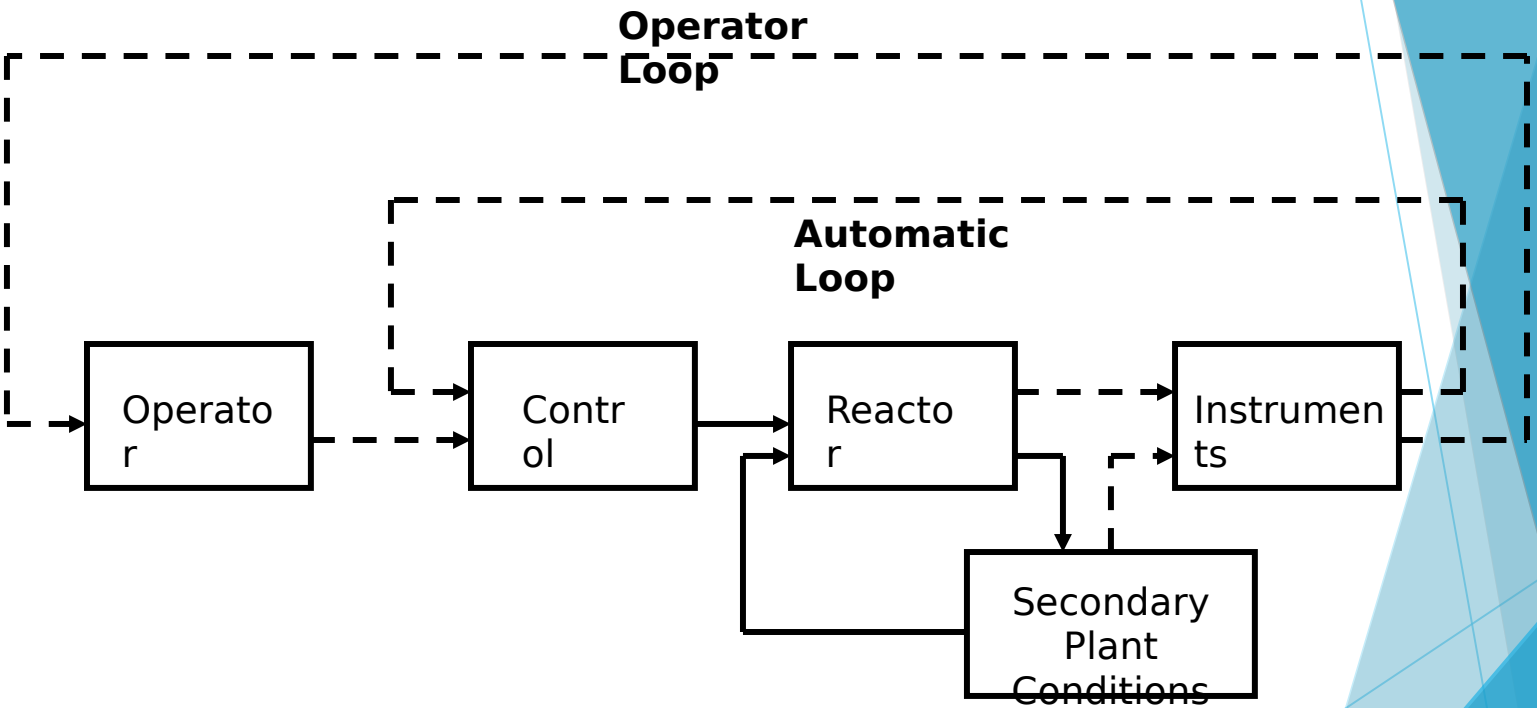
LPRM – local power range monitor
IRM – intermediate range monitor
SRM – source range monitor

Reactor Control System

Responses to changes in neutron flux provide signals to reactor operators as well as to automatic control and safety mechanisms

- Automatic control mechanisms compensate for transient changes in reactivity, such as might result from minor variations in coolant flow, as well as steady changes due to fuel depletion and accumulation of fission product poisons (i.e., neutron absorbing isotopes such as Xe-135 and Sm-149)
- Automatic safety mechanisms terminate abnormal transients or operational upset conditions that exceed control capability of reactor operators and automatic control mechanisms

Reactor Control System Illustration



Reactor Startup

Reactor Startup Neutron Source

- Startup of shutdown reactor requires insertion of neutron source since $k < 1$
- Addition of source neutrons to reactor system with $k < 1$ has effect of maintaining much higher stable neutron population due to fissions occurring than neutron population that would result from source neutrons alone
  - Stable neutron population = $S/(1-k)$, where S is neutron source strength

Reactor Startup Simulation

(Executable file in NuclearReactorSim folder)

Subcritical Reactor Stable Neutron Population Illustration for k = 0.5

Generation	1st	2nd	3rd	4th	5th	6th	7th	8th	9th	10th
Neutrons	100	50	25	12	6	3	1	0	0	0
		100	50	25	12	6	3	1	0	0
			100	50	25	12	6	3	1	0
				100	50	25	12	6	3	1
					100	50	25	12	6	3
						100	50	25	12	6
							100	50	25	12
								100	50	25
									100	50

A neutron source adding 100 neutrons per generation will result in a stable neutron population of ~200 neutrons being produced from a combination of source neutrons and fission in a reactor with a k of 0.5.

Subcritical Multiplication Factor

- For reactor with k < 1, effect of fissions in fuel causes increase in effective neutron source strength – called *subcritical multiplication*
- Subcritical multiplication factor (M) relates reactor steady-state neutron level (N) to neutron source strength (S)
  - Steady-state neutron level is N = S/(1-k), or N = SM,
    where M = 1/(1-k)
  - Note that M is one for shutdown reactor (k = 0) and
    becomes infinite as k approaches one (i.e., reactor approaches critical)

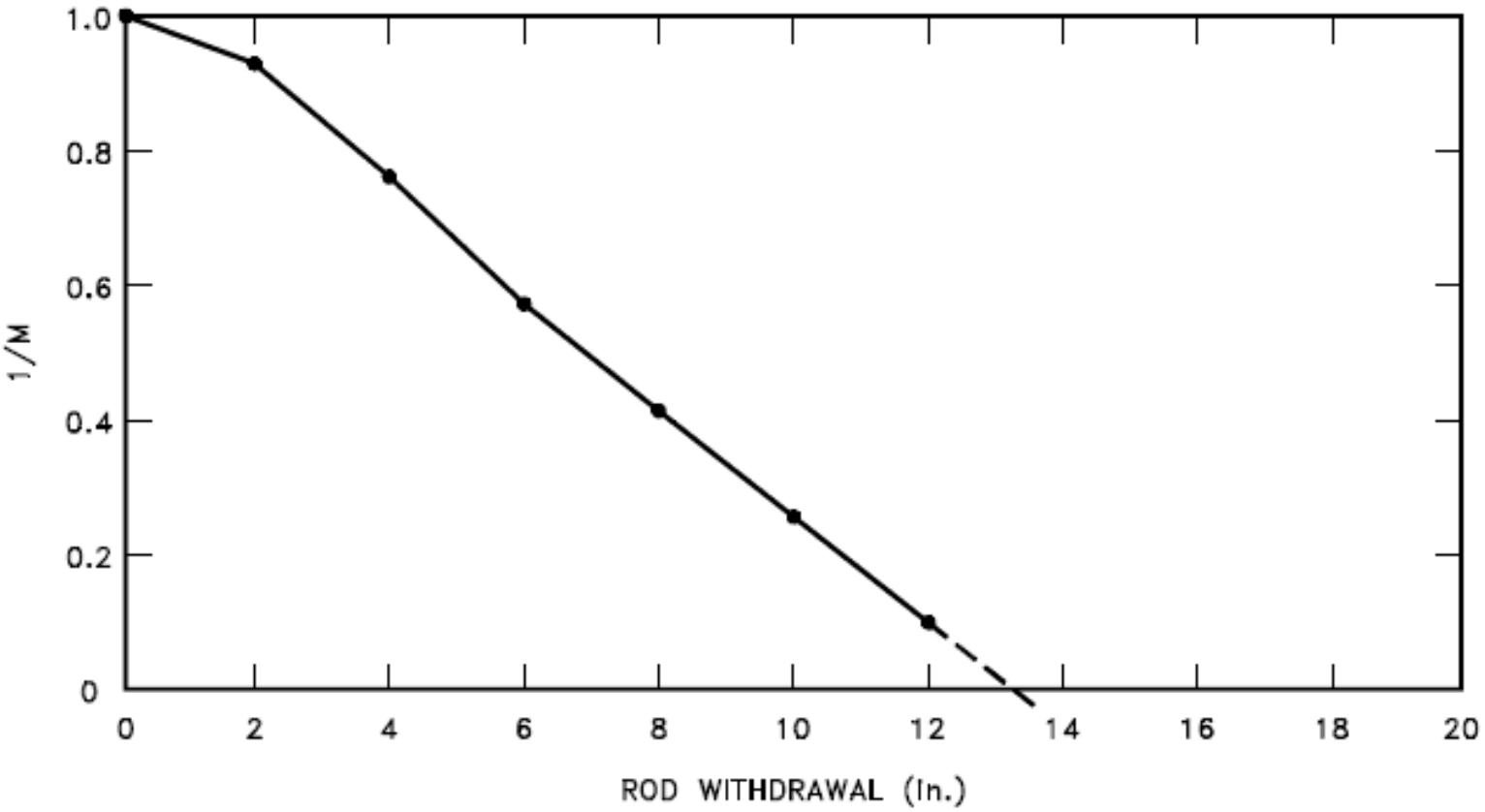
Starting Up Shutdown Reactor

- Inverse of subcritical multiplication factor (1/M) used to monitor approach to reactor criticality since M related to value of k
  - As positive reactivity added to subcritical reactor (e.g., control rods are withdrawn), k gets closer to one (criticality) and 1/M (= 1-k) approaches zero
- Neutron monitoring system instrumentation used in reactor startup procedure to determine how far to withdraw control rod bank from reactor core to achieve criticality (k = 1)

Starting Up Shutdown Reactor (cont.)

- Reference neutron monitoring system count rate, CR_o, taken before any control rod bank movement
- Value of 1/M determined from $1/M = CR_o/CR$ following step withdrawal of control rod bank, where CR is new steady-state neutron count rate following each step change
- 1/M plotted vs. control rod bank withdrawal until value can be accurately extrapolated to zero, which determines amount of control rod bank withdrawal for achieving reactor criticality (next slide)

1/M Plot vs. Control Rod Withdrawal



Terribly Oversimplified Summary

As stated at beginning – it's all about neutrons

- Fission reaction rate regulated by controlling reactor core neutron economy
- Neutron population and rate of population change controlled by proper reactor design and application of detailed operating procedures
- Real-time neutron population monitoring by neutron detection instrumentation provides immediate input to reactor operators as well as automatic control and safety systems

Show of Hands

How many of you believe you now have a basic understanding of how a nuclear reactor works?

Supplemental Slides

Neutron Multiplication Factor

Value of system neutron multiplication factor can be determined from mathematical expression for neutron life-cycle e\ $\varepsilon \mathcal{L}_f p \mathcal{L}_t f \eta$, where:

- **ε** (fast fission factor) and η (reproduction factor) account for neutron production
- $\mathcal{L}_f$ (fast non-leakage probability) $\mathcal{L}_t$ (thermal non-leakage probability), p (resonance escape probability),
  and f (thermal utilization factor) account for neutron losses

Neutron Multiplication Factor (cont.)

- Values of factors in six-factor formula have limited ranges
- Values of neutron production factors depend on amount of fissile material in reactor core (fuel enrichment), size and spacing of fuel rods, and core age
- Values of neutron loss factors depend on fuel enrichment, reactor materials, core dimensions, core age, and reactor power level

Neutron Multiplication Factor (cont.)

Formula Component	Typical Thermal Reactor Value
ε	1.04
$\mathcal{L}_f$	0.87
p	0.8
$\mathcal{L}_t$	0.86
f	0.8
η	≈ 2

Limiting Neutron Multiplication Factor

Maximum value of k can be limited by decreasing thermal utilization factor (f)

- through introducing burnable poisons into reactor
  fuel (e.g., compounds of boron or gadolinium)
- by inserting control rods into reactor core

Response to Increase in BWR Recirculation Flow